Gamma-Ray Effects Testing in
Lawrence Livermore
National Laboratory's
Nova Upgrade Facility

Thesis

David H. Marchant, Captain, USAF

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Gamma ray effects testing in Lawrence Livermore National Laboratory's (LLNL) planned Nova Upgrade facility is examined. Emphasis is placed on converting neutron energy from inertial confinement fusion in to gamma rays while shielding the test objects from neutrons and debris. Although predicted gamma doses in the Nova Upgrade facility are an order of magnitude less than those produced in some current facilities, dose uniformity, the ratio of minimum to maximum gamma dose is predicted to be greater than 0.75 across a larger, 13,000 cm^2, test bed. Peak gamma dose rates are predicted to be on the order of 10^{10} Gy/s, similar to the dose rates of current simulators. Surprisingly, the laser ports reduce the gamma dose about 30% and the peak gamma dose rate about 40%, but they increase the average gamma energy about 20%. The dose and dose rates from the Nova nuclear weapons effects test (NWET) cassette should scale linearly with the yield from future ICF facilities, such as the Laboratory Microfusion Facility (LMF) planned by the Department of Energy.

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Thesis

Presented to the Faculty of the School of Engineering
of the Air Force Institute of Technology
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In Partial Fulfillment of the
Requirements for the Degree of
Master of Science in Nuclear Physics

David H. Marchant, B.S.
Captain, USAF

March 1992

Approved For Public Release; Distribution Unlimited.

Preface

In this study, I examined the possibility of using the future Nova Upgrade laser facility for testing the effects of gamma rays from nuclear weapons. Although, my work is based on the expected Nova specifications, it is applicable to any inertial confinement fusion facility; doses and dose rates can be scaled for different fusion yields.

I would like to acknowledge the guidance of my faculty advisor, Major Denis Beller. I am also indebted to my colleague and friend, Captain Jeffery Malapit, who helped me keep my perspective throughout the research phase. Finally, I thank my wife, Gabriela, and my children, Andrew, Daphne, Brian, and David for their support and patience.

Table of Contents

List of Figures

List of Tables

Abstract

Gamma ray effects testing in Lawrence Livermore National Laboratory's (LLNL) planned Nova Upgrade facility is examined. Emphasis is placed on converting neutron energy from inertial confinement fusion in to gamma rays while shielding the test objects from neutrons and debris. Although predicted gamma doses in the Nova Upgrade facility are an order of magnitude less than those produced in some current facilities, dose uniformity, the ratio of minimum to maximum gamma dose is predicted to be greater than 0.75 across a larger, 13,000 cm^2, test bed. Peak gamma dose rates are predicted to be on the order of 10^{10} Gy/s, similar to the dose rates of current simulators. Surprisingly, the la_er ports reduce the gamma dose about 30% and the peak gamma dose rate about 40%, but they increase the average gamma energy about 20%. The dose and dose rates from the Nova nuclear weapons effects test (NWET) cassette should scale linearly with the yield from future ICF facilities, such as the Laboratory Microfusion Facility (LMF) planned by the Department of Energy.

Gamma-Ray Effects Testing in Lawrence Livermore National Laboratory's Nova Upgrade Facility

I. Introduction

Background

Nova is Lawrence Livermore National Laboratory's (LLNL) high powered laser facility. It is used to conduct inertial confinement fusion (ICF) experiments. In ICF, high-powered laser beams strike a small pellet, containing fusion fuel, tritium and deuterium. The energy imparted by the laser beams vaporizes and expels the pellet's outer layer. As the outer layer material ablates, it transfers kinetic energy, through conservation of momentum, to the remainder of the pellet. If the energy is transferred uniformly it compresses the pellet. If the laser beams are sufficiently powerful and correctly coupled with the pellet to avoid preheating, they compress the pellet until fuel densities reach the point at which fusion can take place (LLNL, 1989:5 and Krane, K., 1988:546-551).

Currently, the Nova laser delivers about 120 kJ of energy in ten beams to an ICF pellet (Krane, K., 1988:551). LLNL is building a 288 beam laser for Nova with an output energy between 1 and 2 MJ. With these changes, LLNL expects to be able to produce fusion yields up to 45 MJ (Tobin, M.,

1991:1). With these yields it may be possible to use the Nova Upgrade facility for nuclear weapons effects testing (NWET). However, the distribution of energy between neutrons and photons from the ICF pellet will be different than that from a thermonuclear bomb. In a nuclear bomb, a large portion of the neutrons consists of high-energy neutrons. Interactions with the heavy elements used in bomb construction convert much of the neutron energy to photons. Photons make up most of the energy released from a thermonuclear bomb (Glasstone, S., 1977:340-342). The neutrons from an ICF pellet are fast neutrons also, but the pellet material is too light and too small to absorb many neutrons. Most of the energy released in ICF is in the kinetic energy of the neutrons. To use the Nova Upgrade facility for gamma-ray effects testing, a method is needed to convert as much energy as possible from the fusion neutrons to gamma rays.

Problem and Scope

One method of converting fusion neutron energy into gamma rays is through inelastic collisions of neutrons with nuclei. When a neutron collides inelastically with an atom, it creates a compound nucleus. The nucleus then expels the neutron. The combined kinetic energy of the expelled neutron and the nucleus together is less than the kinetic energy of the original neutron. When the nucleus falls back to ground state, it releases its excess energy as gamma rays

(Knoll, G., 1989:58). Gamma rays are photons originating from nuclei. They generally have high energies, typically between 100 keV and 10 MeV (Krane, K., 1988:327).

Gamma rays damage systems by ionizing atoms. The amount of damage done to an electronic device due to ionization depends on the rate at which the gamma-ray dose is applied and on the total gamma-ray dose received (Messenger, G., 1986:266). To be able to test the effects of gamma rays on systems, high gamma-ray dose rates and high total gamma-ray doses are required. In this study, I concentrated on generating as high a gamma-ray dose and as high a peak gamma-ray dose rate as possible in an NWET cassette inserted into the Nova Upgraded chamber. Three figures of merit were used: 1) ratio of gamma-ray to neutron dose; 2) total gamma-ray dose; 3) peak gamma-ray dose rate. Concerns associated with gamma-ray production in the Nova Upgrade facility include:

1. Finding materials that efficiently convert energy from fusion neutrons to gamma rays with an average energy above 1 MeV.

2. Finding shape, thickness, and position of shields and converters that maximize the total gamma-ray dose, the peak gamma-ray dose rate, and ratio of gamma-ray dose to neutron dose at the test area.

3. Comparing calculated results with the specifications from current gamma-ray and neutron testing facilities and listing ranges of calculated total gamma-ray doses and gamma-ray dose rates.

Approach

The Oak Ridge National Laboratory (ORNL) Monte Carlo transport code, MORSE, was used to model NWET in the Nova Upgrade. The models included cross sections for each material used, the source spectrum, response functions, and the geometry for each case. Cross-sectional data came from the ORNL's Defense Nuclear Applications Broad-Group Library (DABL69) (Ingersoll, D., 1988:5.1-5.2). The source spectrum consisted entirely of neutrons in the 13.84-MeV to 14.191-MeV energy group (DABL69 energy group 4). The response functions included gamma-ray and neutron kinetic-energy-released-in-material (KERMA) factors for silicon, gamma-ray and neutron fluence factors, and factors for computing the average energy of gamma rays and neutrons. NWET cassette geometry was made up of a neutron-to-gamma converter, neutron and debris shields, and a supporting can. Different converter shapes and materials were modelled and compared to find the configuration that delivered the highest total gamma-ray dose, peak gamma-ray dose rate, and ratio of gamma-ray to neutron dose.

Before comparing results from different materials and cassette configurations, I developed a baseline case with a glass (SiO_2) converter. Each succeeding case represented an evolution of the baseline case. Any change in the results was attributed to the corresponding change in the configuration. The dose, peak dose rate, average energy,

and pulse width for each case were compared against this baseline and against the other cases. A "best" case design was developed by combining materials and shapes that gave high gamma-ray doses, gamma-ray dose rates or ratio of gamma-ray dose to neutron dose 3.5 meters below the ICF source.

The "best" design was then analyzed for gamma-ray dose uniformity, for average gamma-ray energy and for gamma-ray spectral properties. Uniformity was calculated by taking the ratio of the minimum dose to the maximum dose in a dose plane 2 m below the ICF source. To do this, the gamma-ray dose was calculated at five evenly spaced locations: 0, 16, 32, 48 and 64 cm from the can's axis.

Finally, the laser ports were included in the baseline geometry. The gamma-ray dose, peak dose rate, and average energy were compared against the original baseline case. The changes were then applied to the "best" case results. The adjusted total gamma-ray doses and gamma-ray dose rates, were compared to current above ground gamma-ray effects simulators.

Sequence of Presentation

The thesis begins with a brief description of the Monte Carlo computer code used to model the Nova Upgrade facility and the NWET cassette. This is followed by a description of the response functions used to evaluate the results, and descriptions of the Nova Upgrade chamber and a generic NWET

cassette. Descriptions of the different NWET cassettes studies include the baseline study; the study of gamma-ray generating materials; studies of thick converters, reflecting converters, and extended converters; a study of pulse stretching; and a study of the effects of the laser ports on the baseline case. The results for each case are presented and discussed in the same order. Finally, a comparison is given between projected results from the Nova Upgrade facility and current above-ground test facilities.

II. Gamma-Ray Generation in the Nova Upgrade

Description of MORSE

MORSE is Oak Ridge National Laboratory's (ORNL) Monte Carlo computer code for neutron and gamma-ray transport in three dimensions. In MORSE, the user defines geometric objects, materials and their cross sections (in either ANISN or DTF-IV formats), locations in the geometric objects where the results are to be calculated, source particle information, and the response functions. MORSE calculates particle fluence per primary particle, source neutrons, at each detector location for each user specified energy bin, time bin, and angle bin. The fluence is multiplied by the response functions for total responses and for time dependent responses (Emmett, M., 1984:4.1.1-4.2.5).

Response Functions

In this study, three response functions were used for each particle type: the kinetic-energy-released-in-material (KERMA) due-to-ionization response, fluence response, and average energy response. KERMA due to ionization was used because it is the primary damage mechanism for gamma-rays in semiconductor devices (Messenger, G., 1986: 266). The response functions consist of values for each energy group that, when multiplied by the fluence in that energy group, gives the desired detector response. For example, to find

the neutron fluence, a response value of one is used for each neutron energy group and a response of zero is used for each gamma-ray energy group. For each detector MORSE can track uncollided and total responses, energy dependent responses, time dependent responses, energy and time responses, and angle and energy responses. The total detector response is the fluence integrated over the response at a given detector (Emmett M., 1984: 4.6-1).

KERMA. The KERMA factors used in this study came from the DABL69 library (see Appendix A, tables 11 and 12) (Ingersoll, D., 1988:14-17). They represent the kinetic energy deposited in silicon by photoelectric and Compton scattering processes per unit fluence. The KERMA per unit fluence for a particular energy group is defined by the following relationship (Messenger, G., 1986:373-374):

$$K_g = 1.602 \times 10^{-13}\ n\ E_g\ \sigma_g/\rho \text{ (J per g/ unit fluence)} \qquad (1)$$

Where

g is the energy group

E_g is the energy deposited in silicon per incident particle (MeV/particle)

n is the number density of Silicon (atoms per cm^3)

σ_g is the energy group cross section (cm^2/atom)

ρ is the density of silicon (grams per cm^3)

The total dose due to ionizing radiation in a group is the product of the group's KERMA factor and the estimated fluence for the group:

$$\text{Total Detector Response} = K_g \, \phi_g \qquad (2)$$

Where ϕ_g is the differential fluence in each energy group. Since the energy groups are numbered without regard to whether the group represents gamma-ray or neutron energy, the KERMA factors for gamma rays and neutrons are put in separate tables in the MORSE input file (see Appendix C). A value of zero is used for the energy groups not corresponding to the particle type of interest, ie. neutron energy groups in the gamma-ray KERMA table.

Average Energy Responses. The response functions for calculating the average neutron and gamma-ray energies were simply the average energy of each energy group. The arighmetic mean was used because the nature of fluence within the group is unknown. The total detector response is an average-energy weighted fluence. Dividing this average energy weighted fluence by the total fluence gives the average gamma-ray or neutron energy at the detector as follows:

$$\text{Average Energy, } E = \frac{\sum_{g=1}^{G} \phi_g E_g}{\sum_{g=1}^{G} \phi_g} \quad (3)$$

Responses in MORSE are given in terms of source neutrons. To make the results scalable with any fusion neutron source the total doses and dose rates were changed to be in terms of fusion yield. Deuterium and tritium fuse according to the formula

$$^{2}D + {}^{3}T \rightarrow {}^{4}He + {}^{1}n + 17.6 \text{ MeV} \quad (4)$$

The number of neutrons produced per 1 MJ of fusion energy released is 3.547×10^{17} neutrons. Unless otherwise specified, results in this report are given in terms of ICF yield in megajoules by multiplying the dose and dose rates by this number. Note, results do not assume a specific ICF yield nor a specific neutron energy partition. In order to scale results to a specific ICF yield, the energy partition of that yield must be known. In the Nova Upgrade facility, the neutrons are assumed to carry 70%, verses the theoretical 80%, of the total ICF yield (Tobin, M., 1991:3).

Physical Description of the Nova Upgrade Facility

Figure 1 shows the Nova Upgrade test chamber. The outside wall of the test chamber will be a spherical shell of 20-cm-thick lead-borated polyethylene blocks fitted together. It will have an inside radius of 4.05 meters. The inside wall will be a 5-cm-thick spherical shell of aluminum (alloy 5083). A hole in the top of the chamber, 20 cm in diameter, will allow a target inserter to be lowered into place. During normal operations the tip of the inserter, with the ICF pellet attached to the tip, will be positioned at the center of the chamber. The chamber wall will have 288 conical laser ports that allow the laser beams to be focused on the pellet. Thirty-six of these ports will be placed in each of four concentric circles centered on each group of diagnostic ports. These ports will be about 13 cm in diameter at the aluminum wall. In the bottom of the chamber a 1.5 meter diameter hole will be cut and fitted for the NWET cassette (Tobin, M., 1991:3-11). Unless otherwise specified, the computer models described below contain only the Nova chamber walls.

Physical Description of the Baseline NWET Cassette

The NWET cassette will contain four basic sections (see figure 2) as follows: a neutron to gamma-ray converter, a direct neutron shield held in place by shock absorbing struts, a supporting can, and a debris shield. The NWET cassette will be self supporting. Although the cassette

will be joined to the Nova chamber and vacuum sealed, it will be easily removable. A lift beneath the Nova Upgrade chamber will give experimenters access to the test objects and diagnostic equipment.

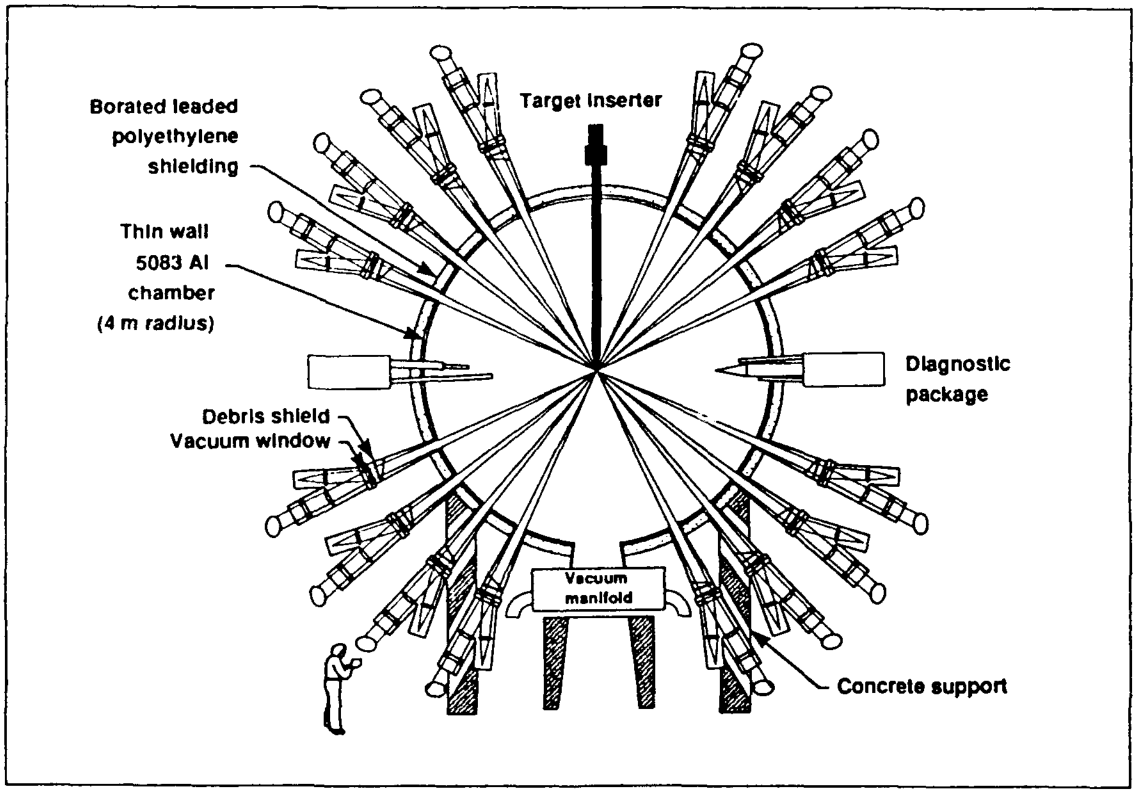

Figure 1. Cross Sectional View of the Proposed Nova Upgrade Test Chamber (Tobin, M., 1991: 4)

The test objects and diagnostics for the NWET tests will be placed in racks in the lower half of the cassette at a distance greater than 2 m from the ICF source. They will be protected from fusion neutrons by a direct neutron shield and from debris by a 0.5-mm thick tungsten or tantalum debris shield. The direct neutron shield is placed 25 cm

below the ICF pellet. It consists of a truncated right cone 18 cm thick, 9 cm of tungsten on top of another 9 cm of borated polyethylene. It is designed to shield the entire inside of the supporting can 2 m from the ICF source. The mass of the direct neutron shield is 47.5 kg, 43.5 kg of Tungsten and 4.0 kg of borated polyethylene. It will be supported on a tripod of shock absorbent struts that will attach to the can wall 2 m from the ICF source. The debris shield will be stretched across the inside of the can 1.2 meters from the ICF source.

The supporting can will be 10 cm thick. This will allow for a 13,000 cm^2 test area. Test objects can be placed within the can below the 2 meter zone protected by the direct neutron shield and the debris shield. The can will contain an aluminum and lead-borated polyethylene plug similar to the Nova chamber wall. This plug will provide radiation shielding for the room directly below the chamber. The NWET cassette will have a vacuum sealed interface with the Nova chamber.

The converter, the direct neutron shield, the debris shield, and the supporting can wall were included in the computer model of the NWET cassette. No attempt was made to model the supporting struts for the direct neutron shield, the target inserter, the diagnostics, or the vacuum manifold. The laser ports were modelled as conical holes, for simplicity, in one case only.

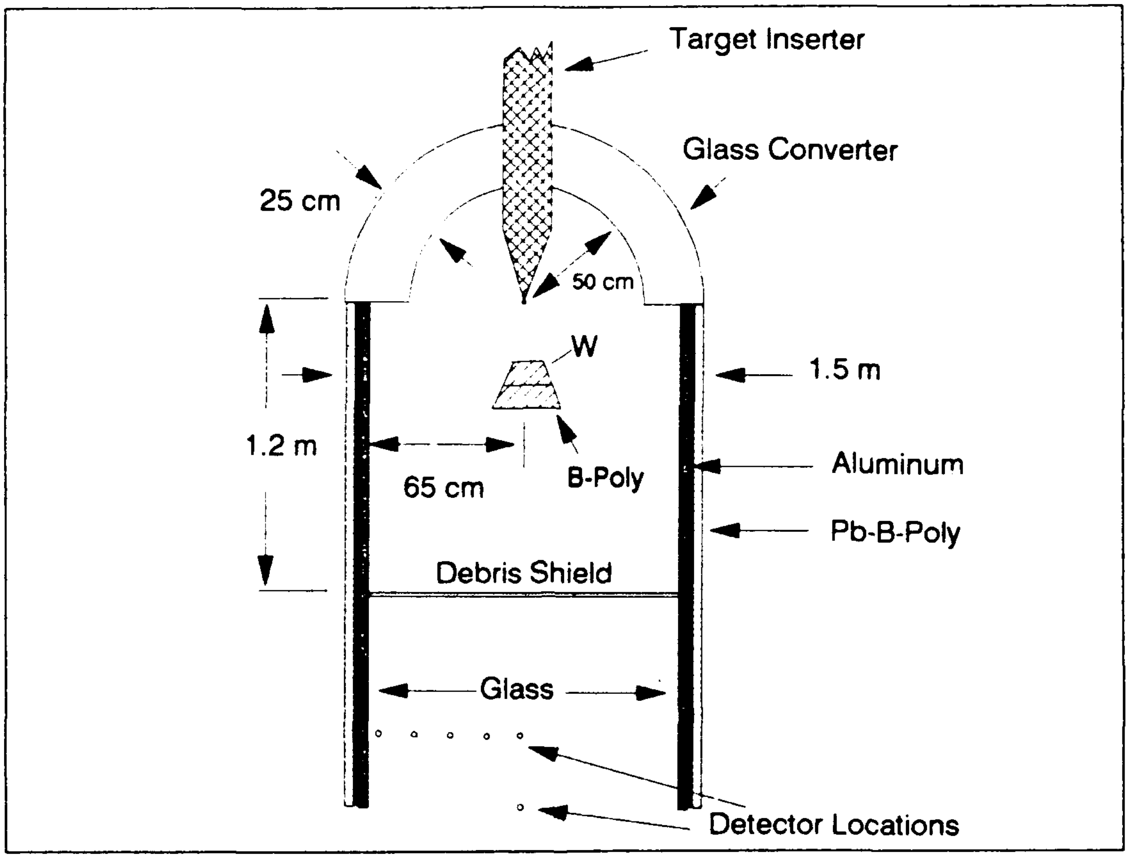

Figure 2. Unscaled Cross Sectional View of the Baseline NWET Cassette Showing Target Inserter, Converter, Supporting Can, Direct Neutron Shield, Debris Shield and Six of the Eight Point Detectors

<u>The Baseline Case</u>

The baseline model served as the blueprint for all the other case studies. The converter, in the baseline case, was a 25-cm-thick hemispherical glass shell, centered on the ICF pellet, with an inner radius of 50 cm. It sat on a 10-cm-thick, 3.929-meter-long, can. The can had a 65 cm inner

radius and was made of 5-cm-thick lead-borated polyethylene on the outside, 2.5-cm-thick aluminum in the middle, and 2.5-cm-thick glass on the inside. Four point detectors were placed along the can's axis at 2, 2.5, 3, and 3.5 meters down from the ICF pellet. Four more point detectors were placed within the can 2 m down from the pellet and 16, 32, 48, and 64 cm radially out from the can's axis (see figure 2). Forty-one time bins ranging from 11 ns to 0.1 second were used to map the dose rate at each detector. The eleven nanoseconds bin corresponds to the earliest time a gamma ray produced in the direct neutron shield could strike the closest detector. The time bins were spaced to give 1 ns resolution between 11 and 60 ns. The lower 66 energy bins were used for the energy spectra.

Study of Gamma-Ray Generating Materials

There are two major issues that dictate whether or not materials make good neutron-to-gamma-ray converters, these are gamma-ray generation and self-shielding. The first is the probability a 14 MeV neutron will inelastically collide in the material and create a gamma ray. To find the best gamma-ray generating materials, the converter must be evaluated by itself. This ensures the total gamma-ray dose is from the converter material. The evaluation was made by modeling twelve different materials in hemispherical converters one mean-free-path (mfp) thick (see Table 1). Note, unless otherwise specified, the mean free paths of 14

MeV neutrons is assumed whenever mean free paths are referred to henceforth. Table 1 also includes the length of one mean free path of 1 to 2 MeV gamma rays in different materials. The average gamma-ray energy in these materials is between 1 and 2 MeV In each case, the detector was placed 3.5 meters from the ICF source opposite the converter. The materials were evaluated according to gamma-ray dose and peak gamma-ray dose rate at the detector. No other part of the Nova chamber or NWET cassette was used.

TABLE 1
Length of One Mean Free Path in Various Materials

Material	Neutrons (cm)	1 - 1.5 Mev Gamma Rays (cm)	1.5 - 2 Mev Gamma Rays (cm)
Phosphorus	8.1	5.3	6.3
Sulfur	28.7	17.2	20.5
Calcium	20.0	11.3	13.4
Iron	4.6	2.4	2.8
Nickel	4.0	2.0	2.4
Copper	4.0	2.1	2.5
Tantalum	3.4	1.1	1.3
90-10 Cupronickel	3.9	2.1	2.5
Lead	5.6	1.5	1.8
Calcium Sulfate	14.0	12.1	14.5
Glass	11.3	10.1	12.1
Aluminum	9.5	6.7	8.0

(Ingersoll, D., 1988:10)

The second important consideration is the probability of gamma rays passing through the material. If the NWET cassette is made of material with large gamma-ray cross

sections more gamma rays will be attenuated in the material. This will result in a lower gamma-ray dose at the detector. On the other hand, neutrons which pass through the NWET cassette strike the Nova Upgrade chamber wall generating a second pulse of gamma rays. If the NWET cassette contains material with small gamma-ray cross sections, the gamma rays produced in the chamber wall may increase the total gamma-ray dose yet decrease the height of the first gamma-ray pulse. To find the effects the wall gamma rays will have for different materials, hemispherical converters 1-mfp thick were evaluated using the baseline geometry. The source neutrons travel at 5.1747×10^7 m/s. At this velocity neutrons strike the Nova chamber wall after 77 ns. Gamma rays generated by these neutrons arrive at the detector closest to the Nova chamber wall after 80 ns. Most of the gamma rays generated in the cassette will have arrived by this time. The ratio of gamma-ray dose deposited before and after 70 ns shows how much of the total gamma-ray dose comes from the converter material and how much from the wall. This is important because, as shown later, the aluminum in the Nova chamber wall is a good gamma-ray generator.

Increasing the converter thickness increases gamma-ray generation and self absorption. The interaction of more neutrons in the material increases gamma-ray generation and total gamma-ray dose. However, the gamma rays created in

many materials have shorter mean free paths than the neutrons creating them. Glass and calcium sulfate are exceptions. Converters, 2.5 mfp thick, were evaluated for glass (the baseline case), for calcium sulfate ($CaSO_4$), for aluminum, and for copper.

Study of Converter Designs

To increase the number of gamma rays incident on the detectors within the NWET can, three ideas were explored: particle reflectors, converters with a large surface area, and gamma-ray generating cans. Gamma rays with energies of interest, 1 to 5 MeV, undergo Compton scattering as their principle interaction with matter. In Compton scatter, the gamma ray interacts with a loosely bound electron. During the process the gamma ray imparts energy to the electron and departs at a new angle described by the Klein-Nishina formula. According to the Klein-Nishina formula, gamma rays, with energies of interest, are more likely to scatter in a forward biased direction. Nevertheless, there is a small probability they will backscatter (Krane, K., 1988:200-201).

Gamma-ray-reflecting converters consisted of a low density gamma-ray generator capped by a high density reflector material. Both materials were 1 mfp thick. However, the reflecting material was many gamma-ray mean-free-paths-thicker than the gamma-ray generating material. Cupronickel and tantalum make good choices for the

reflector. Glass was used as the gamma-ray generator material. It has a small gamma-ray cross section and is a moderately good gamma-ray generator.

In the next two schemes, the surface area of the gamma-ray generating material that the source neutrons strike was increased. First, a 25-cm-thick, 38-cm-long, ring converter was placed under the hemispherical converter. This gave nearly a 3π coverage of the ICF source. Second, the can itself was used as part of the converter. The thickness of the can was kept at 10 cm so no test area was lost. This required a good gamma-ray generator with a short mean free path. Aluminum was used because it is a good gamma-ray generator with a mean free path of about 10 cm. These two schemes were tried separately and together with excellent results.

Study of Pulse Stretching

As previously mentioned, two gamma-ray dose pulses strike the test object. There is a pulse of gamma rays generated in the NWET cassette, and a pulse of gamma rays generated in the Nova chamber wall. To stretch out the first pulse, a converter with a large surface area is required. The effects of an ellipsoidal converter were examined to minimize the discontinuities in the pulse. Glass, aluminum, and iron were used in the ellipsoidal cases.

To stretch out the second pulse, a converter was not used. Instead, the Nova chamber wall was used as the neutron-to-gamma-ray converter. The test area was shielded from direct neutrons by the standard direct neutron shield and from debris by the normal tantalum shield, but the can was removed to allow wall-generated gamma rays to pass through easily.

Study of Laser Port Effects.

The Nova Chamber includes 288 laser ports. Corresponding holes must be cut in the NWET cassette. These holes will decrease both the gamma-ray and the neutron doses in the NWET cassette. The laser ports were modelled as 288 conical holes cutting through the Nova chamber wall and the baseline NWET cassette to predict the effect of the laser port holes on the total dose, the dose rate, the average energy, and the ratio of gamma-ray to neutron dose. The only detector used in this case was located on the can's axis 3.5 m below the ICF source. The percentage of change in results between the baseline case and the baseline-with-holes case was then used to predict the effects of the holes on the results from the "best design" case.

III. Results and Discussion

Baseline Results

The baseline case was used to examine gamma-ray dose uniformity across the can 2 m below the ICF source, to explore the time dependency of the gamma-ray dose, and to serve as standard against which to compare the other cases. The ratio of minimum gamma-ray dose to maximum gamma-ray dose measured across the NWET cassette 2 m below the ICF source is 0.79. There was no statistical difference in total gamma-ray dose in detectors out to 48 cm from the can's axis. The only significant change in the gamma-ray dose appears near the can wall. This excellent gamma-ray dose uniformity gives a usable test area of about 13,000 cm^2.

Figure 3 shows the time dependency of the gamma-ray and neutron doses 3.5 m below the ICF source. Note the two gamma-ray pulses. The first is generated in the NWET cassette and the second is from the Nova chamber wall. Although the pulse from the Nova chamber wall is an order of magnitude less than the pulse from the NWET cassette, it has twice the pulse width. The relatively long mean free path of glass for the gamma rays generated in the chamber wall accounts for the size of the second pulse. The cumulative gamma-ray dose plot in figure 4 shows that the first gamma-ray pulse delivers about 75% of the total gamma-ray dose to

the detector. The amount of total gamma-ray dose coming from the NWET cassette varies, depending on the material in the NWET cassette, between about 70% and 90%. The irregularities in the first pulse correspond to the neutrons and gamma rays scattering off different parts of the NWET cassette, such as the direct neutron shield or supporting can wall. Generally, these dose-rate characteristics are the same for all converter materials and shapes.

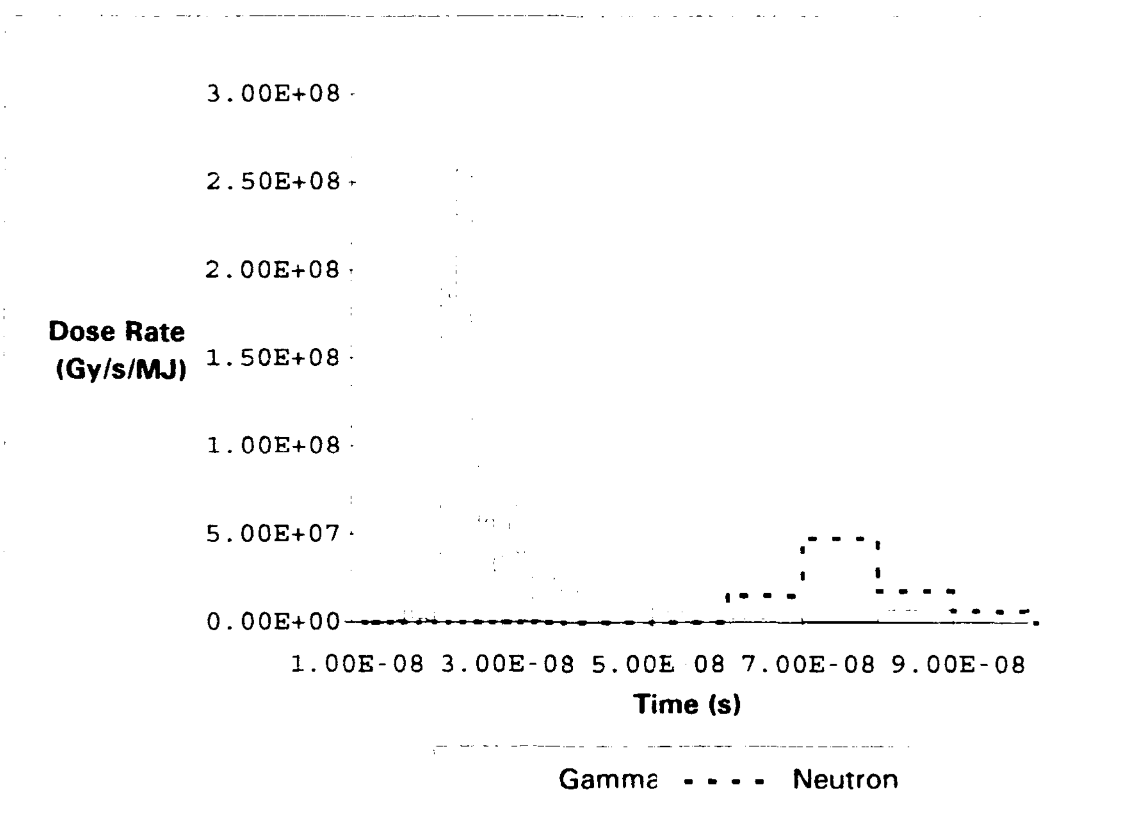

Figure 3. Dose Rate 3.5 Meters From the Baseline Converter

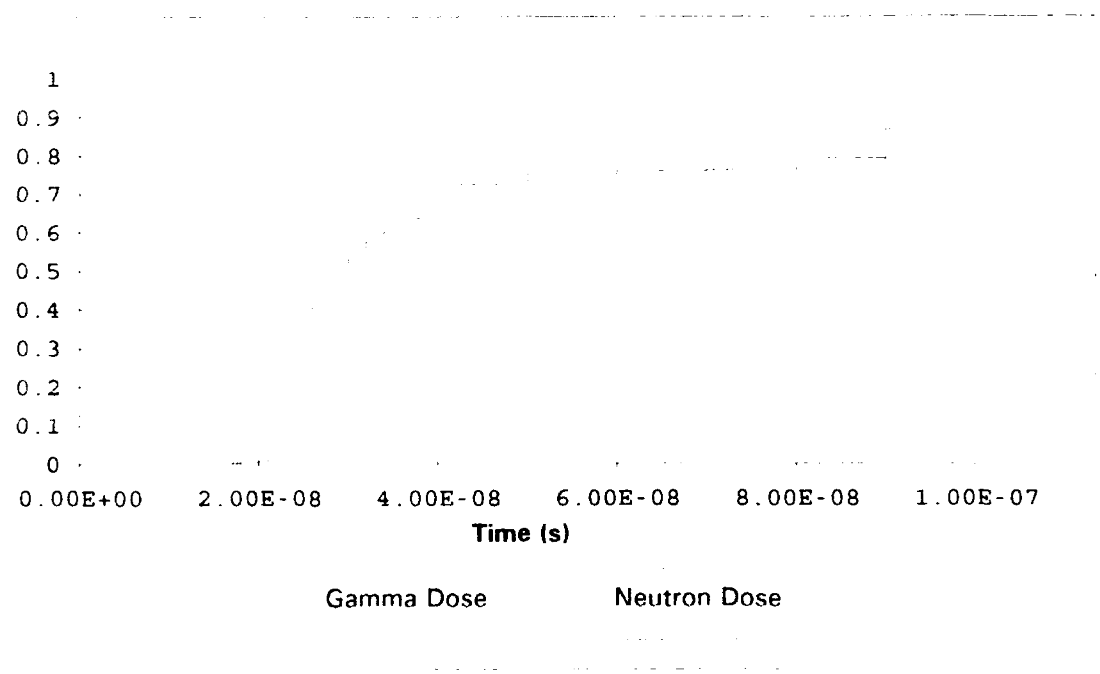

Figure 4. Normalized Cumulative Dose 3.5 Meters from the Baseline Converter

Table 2 lists the results from the baseline case by the detector location. The detector locations are defined by r, the radial distance from the can's axis, and z, the distance from the ICF source along the can's axis. Note, the average gamma-ray energy and the primary pulse's full width at half maximum remain the same throughout the test cassette. No uncertainties less than 1 part in 1000 are reported because MORSE only gives uncertainties greater than or equal to 1 part in 1000. When no uncertainty is given all digits of the value can be considered significant. The results from the fourth detector are compared with the results in the case studies to follow.

Table 2
Results From the Baseline Case

Detector Location (r,z)	Gamma-ray Dose (Gy/MJ)	Peak Gamma-Ray Dose Rate ($\times 10^8$Gy/s/MJ)	Ratio of Gamma-ray to Neutron Dose	Average Gamma-ray Energy (MeV)
(0, -200)	7.83±0.36	7.47	3.65±0.20	1.56±0.11
(0, -250)	4.84±0.21	5.03	2.79±0.14	1.47±0.10
(0, -300)	3.32±0.13	3.26	2.29±0.11	1.45±0.09
(0, -350)	2.52±0.11	2.57	2.17±0.12	1.33±0.10
(16,-200)	7.78±0.41	7.1 ±0.7	3.61±0.22	1.56±0.12
(32,-200)	7.83±0.56	7.81±0.81	3.25±0.28	1.45±0.17
(48,-200)	7.56±0.58	7.66±0.74	2.91±0.29	1.43±0.17
(64,-200)	6.18±0.49	5.111	2.25±0.25	1.48±0.17

The Effect of the Debris Shield on the Gamma-Ray Spectrum

One interesting finding in this baseline study was the effect of the debris shield on the average gamma-ray energy. Figure 5, a spectral analysis with and without the debris shield, tungsten in this case, shows a definite hardening of the gamma-ray spectrum when the debris shield was modelled. The average gamma-ray energy increased from 1.09±0.05 MeV, without the debris shield, to 1.39±0.13 MeV, with the debris shield, while the average neutron energy dropped from 2.01±0.10 MeV to 1.70±0.09 MeV. The impact of this energy shift showed up in the total gamma-ray dose received at 3.5 meters.

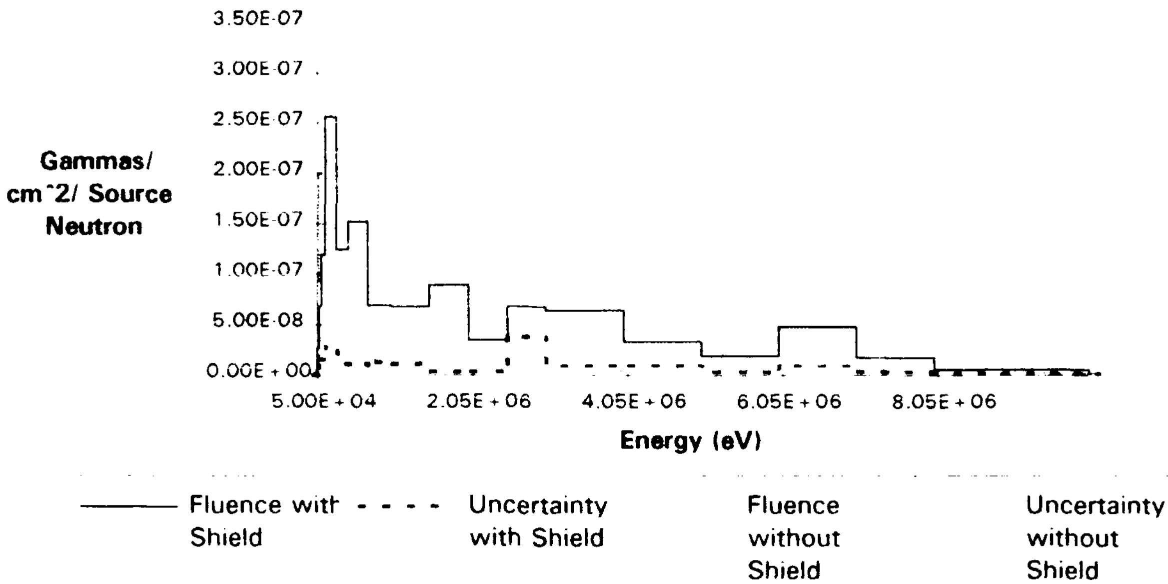

Figure 5. Gamma-Ray Fluence, with Uncertainties, at 3.5 m With and Without 0.5 mm Tungsten Debris Shield

Gamma-Ray Generating Materials

To determine which materials convert neutron energy to gamma-ray energy most efficiently, a hollow hemispheric converter, 1-mfp thick, was modelled by itself. Twelve materials were evaluated in this configuration for their ability to deliver a high total gamma-ray dose, a high peak gamma-ray dose rate and/or high average gamma-ray energies. Results are shown in table 3. Calcium sulfate delivered the highest total gamma-ray dose and copper delivered the highest gamma-ray dose rate. Lighter materials tended to have higher average gamma-ray energies. Note the high

gamma-ray dose and dose rate from the aluminum. This is important because the inner Nova chamber wall is made of aluminum. It suggests that, unless the NWET cassette shields the test objects, a large portion of the gamma-ray dose will be from wall-generated gamma rays. Another important trend is the low gamma-ray doses and dose rates from elements of high atomic number.

TABLE 3

Results From Converter Alone

Material	Total Gamma-Ray Dose (Gy/MJ)	Gamma-Ray Dose Rate ($\times 10^8$ Gy/s/MJ)	Average Gamma-Ray Energy (MeV)
CaSO4	1.567±0.037	1.816	1.97±0.09
Copper	1.344±0.068	2.486	1.20±0.09
Aluminum	1.332±0.092	2.012	1.66±0.16
Phosphorus	1.311±0.038	2.03 ±0.38	1.55±0.06
Cupronickel	1.256±0.096	2.227	1.23±0.12
Iron	1.154±0.041	2.065	1.37±0.06
Sulphur	1.110±0.068	1.447	1.79±0.08
Nickel	1.088±0.062	1.784	1.55±0.14
Glass	0.981±0.072	1.449	1.90±0.19
Calcium	0.923±0.057	1.125	1.85±0.15
Tantalum	0.495±0.070	0.985	1.46±0.29
Lead	0.327±0.038	0.939	1.52±0.24

When the Nova chamber and the rest of the NWET cassette were modelled with 1 mfp of converter material, there was little statistical difference in gamma-ray dose from most of the materials. Table 4 shows the results of adding the rest of the Nova Upgrade geometry. The total gamma-ray dose and

the peak gamma-ray dose rate increased in each case when the rest of the geometry was modelled.

TABLE 4

Results From Converters 1 mfp Thick In the NWET Cassette

Material	Total Gamma-ray Dose (Gy/MJ)	Gamma Dose-ray Rate ($\times 10^8$Gy /sec/MJ)	Average Gamma-ray Energy (MeV)	Gamma-ray to Neutron Dose Ratio
Iron	2.34 ±0.12	2.496	1.05±0.08	2.43±0.19
Copper	2.271±0.008	2.873	1.25±0.10	2.35±0.16
Sulphur	2.18 ±0.14	1.843	1.38±0.13	2.36±0.17
Aluminum	2.04 ±0.13	2.774	1.31±0.12	2.21±0.17
Cupronickel	1.94 ±0.11	3.400	1.06±0.12	2.19±0.13
Calcium	1.89 ±0.13	2.121	1.26±0.15	2.05±0.16
$CaSO_4$	1.88 ±0.16	2.30±0.40	1.48±0.17	1.97±0.20
Glass	1.84 ±0.09	2.501	1.34±0.10	1.94±0.12
Nickel	1.832±0.14	2.582	1.26±0.15	2.08±0.19
Phosphorus	1.79 ±0.12	1.737	1.49±0.17	2.04±0.15
Tantalum	1.66 ±0.12	2.55±0.43	1.08±0.13	1.88±0.14
Baseline	2.52 ±0.11	2.57	1.33±0.10	2.25±0.25

Materials that were poor gamma-ray generators in the study using just the converter, gave much better results when the rest of the geometry was modelled. Gamma-ray generation in the can wall is the likely reason for this. The materials that delivered the highest total gamma-ray doses and highest gamma-ray to neutron dose ratios tended to be of medium atomic mass such as iron and copper. Copper and its alloy, 90-10 cupronickel, delivered the highest gamma-ray dose

rate. Because of the many heavy elements in neutron shields, the average gamma-ray energy decreased for these cases. However, the average energy trend noted above continued. Materials with lower average atomic numbers produced gamma rays with higher average energies.

Few of these materials approach the gamma-ray generation properties of the baseline case. The 1 mfp glass converter case has a total gamma-ray dose about 76% of the baseline. This is not surprising since in the baseline case, the neutrons had a longer path length in which to interact and produce gamma rays. The more gamma rays generated in the converter, the greater the total gamma-ray dose will be.

Thick Hemispherical Converters

The thickness of the converter was increased from 1 mfp to 2.5 mfp by decreasing the inner converter radius. Table 5 shows that for thicker converters, not only the total gamma-ray dose increased but also the ratio of gamma-ray dose to neutron dose increased for the glass and the aluminum cases. The reason for this increase is the fact that, although the added converter volume scatters more neutrons back toward the detector, the neutrons have lower average energies, and, therefore, deposit less energy per neutron in the detector (Knoll, G., 1989:57-58). For example, the average neutron energy for the case using a glass converter 1-mfp thick was 2.06 ± 0.13 MeV while the

average neutron energy of the baseline case (2.5 mfp) was 1.71 ± 0.18 MeV. Even though the gamma-ray and neutron fluences increased about the same amount between the 1 mfp case and the baseline case, the lower energy neutrons cannot convert their energy as efficiently as the gamma rays. Thus the gamma-ray dose increases more than the neutron dose. The same thing happened with aluminum. The average neutron energy went from 1.96±0.13 MeV for the 1 mfp case to 1.573±0.066 MeV for the 2.5 mfp case.

Table 5

Comparison of Results Between Converters 1 Mean Free Path Thick and Converters 2.5 Mean Free Paths Thick

Material	Gamma-ray Dose (Gy/MJ)	Peak Gamma-ray Dose Rate ($\times 10^8$ Gy/s/MJ)	Gamma-ray-to-Neutron Dose Ratio
Glass: 1 mfp	1.84±0.09	2.501	1.94±0.12
Glass: 2.5 mfp (Baseline)	2.52±0.11	2.57	2.17±0.12
Aluminum: 1 mfp	2.04±0.13	2.774	2.21±0.17
Aluminum: 2.5 mfp	2.70±0.22	2.386	2.44±0.24
Copper: 1 mfp	2.27±0.01	2.873	2.35±0.16
Copper: 2.5 mfp	2.92±0.05	4.66 ±0.85	2.35±0.10
$CaSO_4$: 1 mfp	1.88±0.16	2.30 ±0.40	1.94±0.12
$CaSO_4$: 2.5 mfp	2.29±0.16	1.44 ±0.36	2.07±0.17

Table 5 also shows a marked increase in the peak gamma-ray dose rate for copper while the gamma-ray dose rates for the other materials decrease. The pulse peak for the copper converter arrives between 25 and 26 ns after the ICF ignition while the pulse from the glass converter arrives between 23 and 24 ns. This seems to indicate the gamma rays are coming from both the 10-cm-thick copper converter and the can wall. This shown in the energy spectrum, at the time of the peak dose, for the hemispherical copper converter, figure 7, showing the aluminum peaks. Note, from table 1, one mfp of glass, aluminum, and calcium sulfate is about 10 cm. The gamma-ray dose pulse for the glass converters arrives 2 ns later in the thick converter than in the thin converter and is 1 ns wider. This indicates the decrease in peak gamma-ray dose rate from the other converters is caused by the fact that the gamma-ray dose from the can no longer coincides with the gamma-ray dose from the converters.

Reflecting Converters

The use of high density materials as reflectors had a small effect on the total gamma-ray dose and the peak gamma-ray dose rate. Table 6 shows the gamma-ray dose, the peak gamma-ray dose rate, and the ratio between gamma-ray dose and neutron dose. The gamma-ray dose increased 11% compared to the baseline case when cupronickel was used as a

reflector, however, there was no statistical difference in the total gamma-ray dose when tantalum was used. There was a small increase in the gamma-ray-to-neutron dose ratio caused by a marked decrease in average neutron energy. The average neutron energy decreased from about 2.0 MeV to 1.7 MeV for the cupronickel reflector and to 1.6 MeV for the tantalum reflector. The average gamma-ray energy remained about the same in all three cases: 1.3 MeV.

Table 6

Gamma-Ray and Neutron Results 3.5 m from Reflecting Converters

Reflector Material	Gamma-Ray Dose (Gy/MJ)	Peak Gamma-Ray Dose Rate ($\times 10^8$ Gy/s-MJ)	Gamma-Ray-to-Neutron Dose Ratio
Tantalum	1.99 ±0.10	2.554	2.10 ±0.15
Cupronickel	2.049±0.060	2.517	2.336±0.095
Baseline	2.52±0.11	2.57	2.17 ±0.12

Extending the Converter

The extended converter consisted of a 2.5-mfp thick hemispherical converter on top of a 2.5-mfp thick converting ring made of the same materia' '8 cm high as shown in figure 6. This extended converter was used in two cases: one with a baseline can, and one with a gamma-ray generating can.

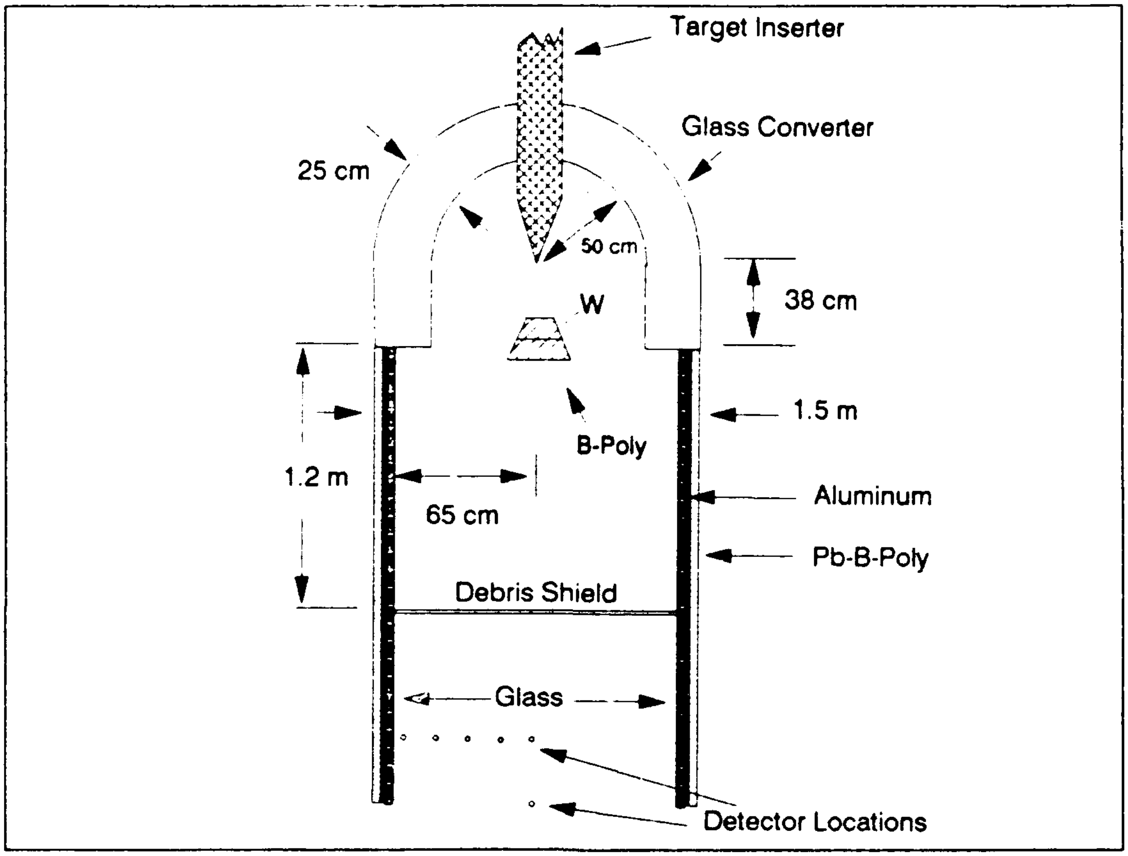

Figure 6. Unscaled Cross Sectional View of the NWET Cassette With the Extended Cassette.

The results from the extended converter on the baseline can are given in table 7. The baseline can was designed to shield the test area from neutrons scattering off the Nova chamber wall. In this case, the only improvement over the baseline case was the peak gamma-ray dose rate for the aluminum converter. The peak gamma-ray dose rate from the copper converter decreased significantly compared to the peak gamma-ray dose rate from the hemispherical copper converter. This is due to the loss of gamma-ray dose from the aluminum in the can wall. Figure 7 shows a comparison

between the energy spectra of the peak pulse from the hemispheric copper converter and the extended copper converter. Gamma rays generated from 14 MeV neutrons in copper are most likely to have energies between 450 KeV and 1 MeV, and between 3 and 7 MeV. Gamma rays generated in aluminum are more likely to have energies between 1 and 4 MeV. The lack of gamma rays with energies between 1.5 and 2.5 MeV for the extended converter indicates few gamma rays are being generated in the aluminum part of the can wall in the extended converter case. It is the lack of aluminum gamma rays that contributes to the lower peak gamma-ray dose rate in the extended converter case. Total gamma-ray dose and the gamma-ray-to-neutron dose ratio remained about the same.

Table 7

Results 3.5 Meters From Extended Converters on Baseline Cans

Converter Material	Total Gamma-Ray Dose (Gy/MJ)	Peak Gamma-Ray Dose Rate ($\times 10^8$Gy/s/MJ)	Gamma-Ray-to-Neutron Dose Ratio
Glass	2.149 ± 0.058	2.392	2.184 ± 0.071
Aluminum	2.57 ± 0.21	3.16	2.71 ± 0.24
Copper	2.90 ± 0.09	3.374	2.18 ± 0.19
Baseline	2.52 ± 0.11	2.57	2.17 ± 0.12

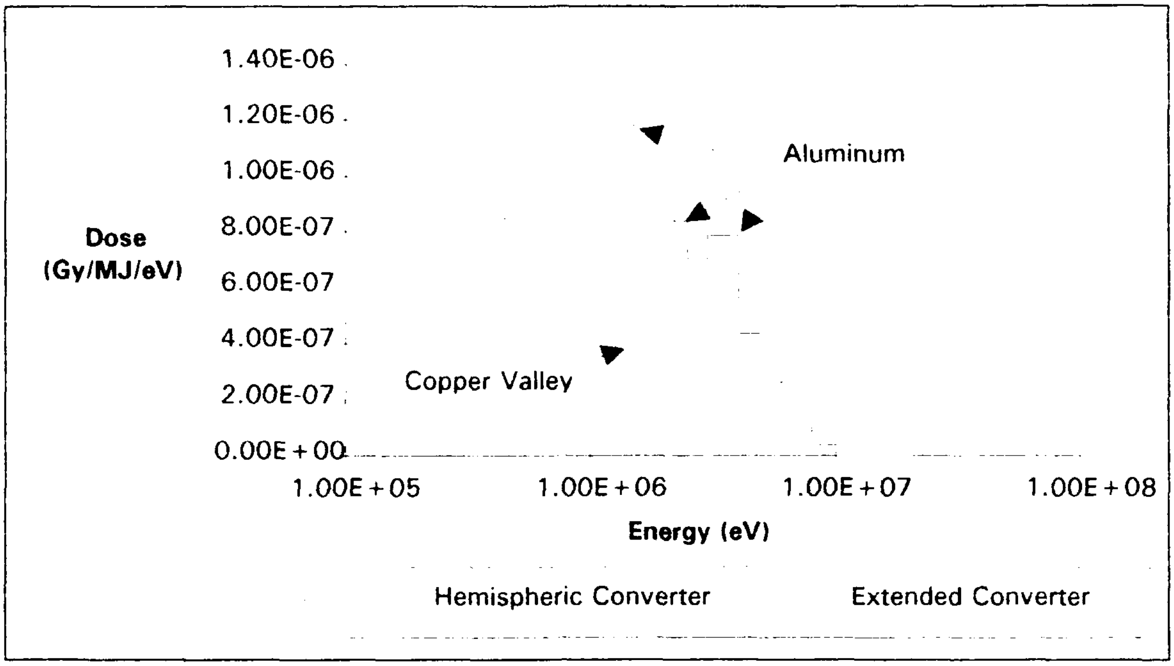

Figure 7 Gamma-Ray Energy Spectra from the Extended and Nonextended Copper Converters on Aluminum Cans

In the second case, the extended converter was placed on an aluminum can. Results are presented in table 8. This improved the total gamma-ray dose for the glass converter more than 10% over the baseline case. The gamma-ray to neutron dose ratio for the glass converter increased about 30% over the baseline case. The most dramatic improvement was with the aluminum converter. Using the aluminum can increased the ratio of gamma-ray to neutron dose by shielding the cassette from neutrons scattered off the Nova chamber wall. Using the gamma-ray-generating can gave the highest total gamma-ray dose, using a copper converter, and the highest ratio of gamma-ray dose to neutron doses, using an aluminum converter.

Table 8

Results From an Extended Converter on an Aluminum Can

Converter Material	Total Gamma-Ray Dose (Gy/MJ)	Peak Gamma-Ray Dose Rate ($\times 10^8$Gy/s/MJ)	Gamma-Ray-to-Neutron Dose Ratio
Glass	2.736± 0.079	2.0	2.90 ±0.13
Aluminum	2.871± 0.067	3.373	3.35 ±0.14
Copper	3.26 ± 0.22	3.58±0.51	2.90 ±0.24
Baseline	2.52 ± 0.11	2.57	2.17 ±0.12

The "Best" Design

Based on the ratio of gamma-ray dose to neutron dose, the extended aluminum converter on the aluminum can was chosen the "best" NWET cassette. This choice of materials is not necessarily optimum. It ignores the consequences of aluminum activation. Also, note that the copper converter case delivers a higher gamma-ray dose but a lower gamma-ray to neutron dose ratio.

The total gamma dose, the peak gamma-ray dose rate, and the gamma-ray-to-neutron dose ratio for a detector 3.5 meters from the source has already been given in Table 8. The full width at half maximum of the converter pulse was 3 ns. Figure 8 compares the gamma-ray pulse of the "best-design" case to the gamma-ray pulse of the baseline case. Table 9 shows the total gamma-ray dose, the gamma-ray dose

rate, ratio of gamma-ray dose to neutron dose, and average gamma-ray energy at different locations in the test region.

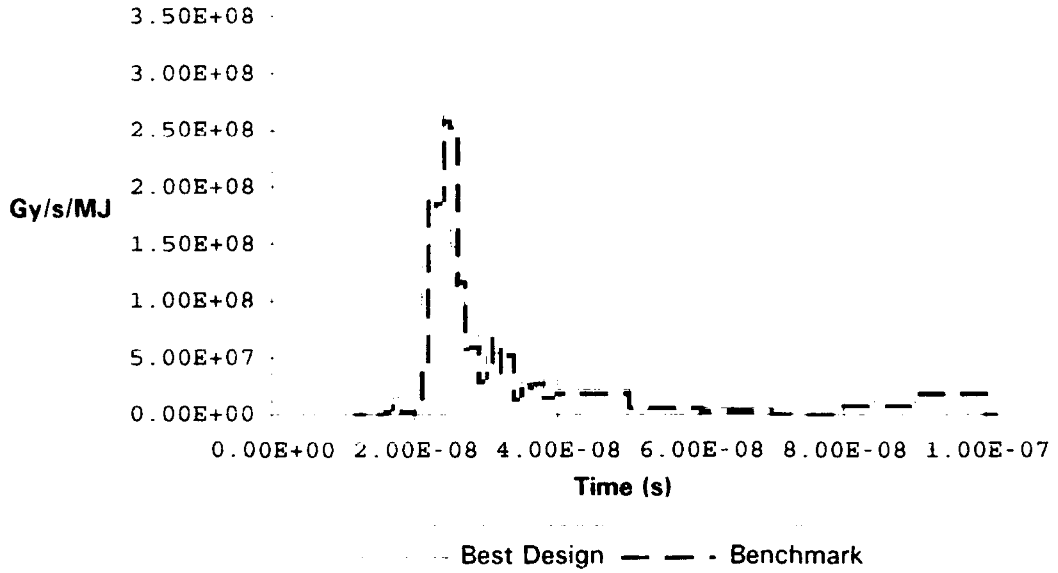

Figure 8. Comparison of Gamma-ray Dose Rates, at 350 cm From Source, Between the Baseline and the Best Design Cases

The uniformity in the "best" case design is better than the uniformity in the baseline case. The ratio of minimum to maximum gamma-ray dose in the detectors at 2 m from the ICF source is about 0.9. The ratio of minimum to maximum peak gamma-ray dose rates is 0.8. Also, the ratio gamma-ray dose to neutron dose drops off as the detectors approach the can wall. This is because of an increase in the neutron dose near the can wall.

Table 9
Results From the Best Design Case at Various Locations

Detector Location (r,z)	Gamma-ray Dose (Gy/MJ)	Gamma-ray Dose Rate ($\times 10^8$ Gy /s/MJ)	Gamma-Ray-to-Neutron Dose Ratio	Average Gamma-Ray Energy (MeV)
(0, -200)	10.60±0.19	11.287	5.63±0.22	1.25±0.03
(0, -250)	6.43±0.18	6.587	4.74±0.17	1.41±0.06
(0, -300)	4.22±0.14	4.586	3.67±0.15	1.28±0.07
(0, -350)	2.87±0.07	3.373	3.35±0.14	1.22±0.04
(16,-200)	10.73±0.39	9.910	6.00±0.36	1.27±0.06
(32,-200)	10.35±0.37	9.510	5.64±0.29	1.23±0.06
(48,-200)	10.13±0.55	9.073	5.36±0.49	1.23±0.09
(64,-200)	9.44±0.53	9.474	4.68±0.64	1.23±0.11

As previously mentioned, aluminum activation may make the "best" case impractical. However, results from most materials are the same order of magnitude. The baseline case, for example delivers a gamma-ray dose up to 110 Gy, a peak gamma-ray dose rate up to 10^{10} Gy/MJ-s, and an average gamma-ray energy of 1.6 MeV at 200 cm from the ICF source.

<u>Ranges For Nuclear Weapon Effects Testing</u>

The ranges for total gamma-ray dose, peak gamma-ray dose rate, and ratio of gamma-ray dose to neutron dose are affected by the configuration of the NWET cassette and by detector location. At 350 cm from the ICF source, the highest peak gamma-ray dose rate, $4.657\times10^{8}\pm0.85$ Gy/s/MJ (4.657×10^{10} Rad(Si)/s/MJ), came from using the 10-cm-thick hemispherical copper converter with the baseline can and the

highest gamma-ray dose, 3.24±0.22 Gy/MJ (324 Rad(Si)/MJ), came from the copper converter on the aluminum can. The "best" design case gave the largest gamma-ray dose with respect to neutron dose. Total gamma-ray dose, peak gamma-ray dose rate, pulse width and ratio of gamma-ray dose to neutron dose are increased by moving closer to the direct neutron shield. They are also increased by increasing the ICF yield. For example, the maximum total gamma-ray dose for the extended aluminum converter and aluminum can increases to about 148 Gy (14.8 kRad(Si)) at 2 m from a 20 MJ source.

Pulse Stretching

Attempts at pulse stretching proved fruitless for the most part. The results from the ellipsoidal glass converter 1-mfp thick were similar to those from the 1-mfp hemispherical glass case. The FWHM increased only 1 ns. The results from the 2.5-mfp thick glass ellipsoidal converter was about the same as those from the 2.5-mfp hemispherical glass converter case. Again, the FWHM increased only 1 ns. The only scheme that showed any real promise of stretching the gamma-ray pulse was to use only the direct neutron shield in the cassette. The Nova chamber wall was used as the converter. This gave a FWHM of 8 ns at 3.5 m but the gamma-ray dose at this distance was only

1.60 ± 0.04 Gy/MJ and the peak gamma-ray dose rate was only 9×10^7 Gy/s-MJ, about half the baseline value.

Effects of Laser Ports

Modeling the baseline case with the laser ports gave surprising results. The total gamma-ray dose with the holes decreased 30% ± 6% from the total gamma-ray dose without the holes. The neutron dose decreased 41% ± 2%. This meant the ratio of gamma-ray dose to neutron dose increased 19% ± 2%. There was a 20% ± 2% change in the average gamma-ray or neutron energies, The peak gamma-ray dose rate decreased 30% ± 2%, and the FWHM of the gamma-ray pulse decreased from 6 to 2 ns.

If the volume a laser port cuts out of the converter is approximated by a right truncated cone, the total volume of the 144 holes in the converter is given by

$$V = 144\ \pi\ (r_1^2 + r_1\ r_2 + r_2^2)\ h\ /\ 3 \tag{5}$$

where r_1 is the radius of the laser port on the inside of the converter, r_2 is the radius of the laser port on the outside of the converter and h is the thickness of the converter. In the baseline case,

$$r_1 = 50 \tan(\theta) = 0.7855 \text{ cm} \tag{6}$$

$$r_2 = 75 \tan(\theta) = 1.1782 \text{ cm} \tag{7}$$

and

$$V = 2.21\text{x}10^4 \text{ cm}^3 \qquad (9)$$

where θ is the half angle of the laser beams (0.9 degrees) (Tobin, M.,1991:5). This represents about 3.6 percent of the total converter volume without the holes. It is surprising that such little volume loss, would result in such a large total dose loss.

Although there is no reason to believe the changes caused by the laser ports are material independent, the basic trends should hold. There should be large decreases in total gamma-ray doses, peak gamma-ray dose rates, and neutron doses, small increases in gamma-ray-to-neutron dose ratio, and little or no change in the average gamma-ray and neutron energies. Table 10 shows the predicted results if the reductions from the baseline holes case are applied to best design case, at 3.5 meters.

Table 10

Predicted Results of Laser Ports in the "Best" Case Configuration

Total gamma dose	2.0 ± 0.2 Gy/MJ
Gamma dose rate	2.0×10^8 Gy/s/MJ
Gamma to neutron ratio	4.4 ± 0.3
Average gamma energy	1.5 ± 0.2 MeV

Comparison With Current Above Ground NWET Facilities

While the Nova Upgrade facility is predicted to deliver a much lower total gamma-ray dose than some current above ground testing facilities, it delivers peak gamma-ray dose rates that is on the same order of magnitude as most current facilities. Current gamma-ray testing facilities include linear accelerators, high voltage flash x-ray machines, and radioactive isotopes (Messenger, G., 1986:218-219). Examples of each of these types are compared to Nova Upgrade in Table 11. Brookhaven is a linear accelerator. Aurora is a flash x-ray facility capable of producing photons of gamma-ray energies. Hermes III is a linear induction accelerator that uses bremsstrahlung to create gamma rays (Choate, L., 1990:25). The Nova Upgrade results are for a 20 MJ fusion yield with 70% of the energy in the neutrons and the "best design" NWET cassette with the test object on the can's axis 200 cm from the ICF source. The results also include the baseline corrections for the laser ports. The criterion given by Choate et al. for gamma-ray dose uniformity in Hermes III was that the ratio of minimum gamma-ray dose to maximum gamma-ray dose be greater than 0.5 (Choate, L., 1990:25). Applying this criterion to the baseline case gives a uniformity greater than 0.85 over 13,000 cm^2. It is clear from table 11 the Nova Upgrade will

be able to meet or exceed most capabilities of present above ground test facilities.

Table 11

Comparison of Capabilities: Nova vs Current Facilities

Facility	Test Area (cm^2)	Maximum Gamma-Ray Dose (Gy)	Maximum Gamma-Ray Dose Rate (Gy/sec)
Nova Upgrade (20MJ source)	13000[a]	114	9.3×10^9
Brookhaven LINAC [b]	10	50000	3.0×10^7
Hermes III[c]	500	>1000	$>5.0 \times 10^{10}$
Aurora[d]	200	5000	1.0×10^{10}
	10000	45	2.5×10^9
^{60}Co[e]	5	N/A	10

[a] $D_{min}/D_{max} > 0.85$

[b] (Ward, T., 1988:68,69)

[c] (Choate, L., 1990:25)

[d] (Davis, J., 1991:7)

[e] (Messenger, G., 1986:219)

IV. Summary and Recommendations

Summary

As a part of its ongoing fusion research, LLNL is upgrading their high powered laser facility, Nova. LLNL expects to achieve controlled ICF ignition with yields up to 45 MJ. In this study I explored using the upgraded Nova facility for gamma-ray effects testing. Figures of merit included the total gamma-ray dose, the peak gamma-ray dose rate, and the ratio of gamma-ray dose to neutron dose.

The proposed NWET cassette is a silo shaped device 5 m high 1.5 m in diameter that can be inserted into the spherical Nova chamber. It consists of a neutron-to-gamma-ray-converting hemispheric shell on top of a supporting can, a conical fusion-neutron shield, and a thin debris shield. The Nova chamber and NWET cassette were modelled using MORSE, a Monte Carlo radiation transport computer code. A baseline cassette using a hemispherical glass converter was evaluated at 8 different locations within the test area of the cassette. The baseline results at 3.5 m from the ICF source were as follows: the gamma-ray dose was 2.52 ± 0.11 Gy/MJ, the peak gamma-ray dose rate was 2.57 Gy/MJ-s, and the ratio of gamma-ray dose to neutron dose was 1.33 ± 0.10. Different shapes and material were examined for the converter and for the supporting can. The results were compared to the baseline find the best configuration.

Although no best configuration was found, trends were found that improved total gamma-ray dose and peak gamma-ray dose rate while increasing the ratio of gamma-ray dose to neutron dose.

Medium Z converters tended to produce higher gamma-ray doses, gamma-ray dose rates, and gamma-ray-to-neutron-dose ratios. In specific, iron, copper, 90-10 cupronickel, and aluminum worked well. All produced gamma-ray doses around 2 Gy/MJ, peak gamma-ray dose rates about 2 $\times 10^8$ Gy/MJ-s, and gamma-ray to neutron dose ratios of about 2.2. Increasing the thickness of the converter from 1 mfp to 2.5 mfps increased the total gamma-ray dose and the gamma-ray to neutron dose ratio. The peak gamma-ray dose rate for the thicker converters dropped in some cases because of the distance between the inner converter radius and the inner can wall. The highest gamma-ray dose rates come from converters that have the same inner radius as the can wall. Combining high density reflectors with a glass gamma-ray generator improved the gamma-ray dose rate about 20% over the baseline case.

The use of extended converters improved total gamma-ray dose and gamma-ray to neutron dose ratios. When aluminum converters were used on a baseline can, the peak gamma-ray dose rate increased about 10% over the baseline case. When the aluminum can was used, the total gamma-ray dose was higher than when the baseline can was used with the same

extended converter. The peak gamma-ray dose rates, increased about 6% for extended copper and aluminum converters, and decreased about 20% for glass converters.

No one case was clearly better than all other cases in delivering the highest gamma-ray dose, the highest peak gamma-ray dose rate and the highest ratio of gamma-ray to neutron doses. The case using the extended aluminum converter on an aluminum can was chosen as the "best design" because of its higher total gamma-ray dose, 2.87 ± 0.07 Gy/MJ, and higher ratio of gamma-ray dose to neutron dose, 3.35 ± 0.14. The highest peak gamma-ray dose rate was generated by the hemispherical copper converter on a baseline can, 4.66 ± 0.85 Gy/s/MJ. The total gamma-ray dose for each of these cases was the same. The maximum pulse width was achieved by removing the neutron-to-gamma-ray converter and the supporting can from the NWET cassette. The FWHM for this case was 9 ns. This pulse stretching comes at the expense of total gamma-ray dose and gamma-ray dose rate.

Finally, the laser ports were added to the baseline model. Although the holes make up less than 5% of the converter volume, they decrease the total gamma-ray dose 30% and the gamma-ray dose rate 40%. However, they increase the gamma-ray to neutron dose ratio by removing much of the neutron dose.

Even with much of the gamma-ray dose removed by the laser ports, the peak gamma-ray dose rate is the same order of magnitude as current capabilities. The total gamma-ray dose deposited in a test object 2 meters from a 20 MJ ICF source is about 114 Gy. This is comparable to many current bremsstrahlung and radioisotope nuclear effects simulators. Uniformity greater than 0.80 over a 13000 cm^2 test bed will make the Nova Upgrade an ideal NWET facility for moderate sized systems and components.

Recommendations

Radiological Hazard. No effort was made to quantify the radiological hazard from neutron capture. Activation of the aluminum, the tungsten and the copper may require an impracticably long cool off time. At a minimum, the NWET cassette should conform to Nova Upgrade safety criteria (Tobin, M., 1991:2).

Further Exploration of Medium Z Materials. The number of materials used in this study was far from exhaustive. Further work might explore the use of P_3S_4 in the "best design" configuration. Use of copper in the inside of the can wall also might give good results. Also, the reflector cases should be examined more closely to find the optimum thickness for different converter materials.

Spherical Coverage. One way of further reducing the neutron dose and increasing the pulse width might be to use a spherical converter or an inverted hemispherical

converter. Materials with large neutron cross sections and small gamma-ray cross sections would have to be used to avoid self shielding problems. Coverter thickness would have to be optimized also.

Appendix A: Gamma-ray and Neutron KERMA Factors for Each Energy Group

There are many different kinds of KERMA factors. The KERMA factors used in this study were those due to ionization. The following tables list the neutron and gamma-ray KERMA factors used in all models:

Table 12. Neutron KERMA Factors

GROUP	Upper Energy (eV)	Neutron KERMA (J/kg-n/cm^2)
1	1.96E+07	1.41E-11
2	1.69E+07	1.19E-11
3	1.49E+07	1.14E-11
4	1.42E+07	1.12E-11
5	1.38E+07	1.11E-11
6	1.28E+07	1.09E-11
7	1.22E+07	1.06E-11
8	1.11E+07	9.66E-12
9	1.00E+07	8.26E-12
10	9.00E+06	7.19E-12
11	8.20E+06	6.42E-12
12	7.40E+06	3.94E-12
13	6.40E+06	1.73E-12
14	5.00E+06	1.30E-12
15	4.70E+06	9.94E-13
16	4.10E+06	6.44E-13
17	3.00E+06	5.83E-13
18	2.40E+06	4.94E-13
19	2.30E+06	5.03E-13
20	1.80E+06	4.09E-13
21	1.42E+06	2.21E-13
22	1.10E+06	2.29E-13

23	9.62E+05	2.60E-13
24	8.21E+05	2.52E-13
25	7.43E+05	1.46E-13
26	6.39E+05	1.94E-13
27	5.50E+05	1.23E-13
28	3.69E+05	1.02E-13
29	2.47E+05	1.11E-13
30	1.60E+05	5.38E-15
31	1.10E+05	7.60E-15
32	5.20E+04	4.08E-15
33	3.43E+04	2.51E-15
34	2.50E+04	1.93E-15
35	2.19E+04	1.31E-15
36	1.00E+04	5.71E-16
37	3.40E+03	1.75E-16
38	1.20E+03	6.56E-17
39	5.80E+02	2.75E-17
40	2.75E+02	3.19E-17
41	1.00E+02	1.59E-17
42	2.90E+01	5.87E-18
43	1.10E+01	2.51E-18
44	3.10E+00	2.79E-18
45	1.10E+00	7.56E-18
46	4.14E-01	8.86E-18

(Ingersoll, D., 1988:13)

Table 13. Gamma-Ray KERMA Factors

GROUP	Upper Energy (eV)	Gamma-Ray KERMA ($J/kg\text{-}photon/cm^2$)
47	2.00E+07	8.50E-12
48	1.40E+07	1.58E-11
49	1.20E+07	2.43E-11
50	1.00E+07	1.94E-11
51	8.00E+06	2.49E-11
52	7 00E+06	2.18E-11
53	6.00E+06	1.86E-11
54	5.00E+06	1.55E-11
55	4.00E+06	1.22E-11
56	3.00E+06	9.80E-12
57	2.50E+06	8.28E-12
58	2.00E+06	6.72E-12
59	1.50E+06	5.07E-12
60	1.00E+06	3.68E-12
61	7.00E+05	2.66E-12
62	4.50E+05	1.84E-12
63	3.00E+05	1.12E-12
64	1.50E+05	7.45E-13
65	1.00E+05	8.36E-13
66	7.00E+04	1.45E-12
67	4.50E+04	3.21E-12
68	3.00E+04	7.39E-12
69	2.00E+04	2.26E-11

(Ingersoll, D., 1988:13)

Appendix B: Material Compositions

Media used in a particular run of MORSE CG is defined in the input file to cross section mixing part of the code: XCHEKR. For XCHEKR to mix media cross sections, it needs number densities for each element in the material. The number densities were calculated from the natural density of each material at standard temperature and pressure assuming homogeneous media. The composition of lead-boron polyethylene used in this report is proprietary but may be obtained by contacting LLNL. The composition of compounds and alloys used in Nova Upgrade models is given in Table 13. Tungsten is included in table 13 because the DABL69 library contains cross sectional data for four stable isotopes of natural tungsten. Natural abundances are used for all isotopes.

Table 14. Compositions of Various Materials

Material	Element	Number Density (Number/barn/cm)
Glass	Si	1.67E-02
	O	3.59E-02
1% Boron Polyethylene	H	7.94E-02
	B-10	6.30E-04
	B-11	2.52E-03
	C	3.97E-02
Al (5083)*	Al	5.64E-02
	Mn	4.17E-04
	Mg	2.62E-03
	Cr	8.93E-05

Tungsten	W-182	1.67E-02
	W-183	9.06E-03
	W-184	1.94E-02
	W-186	1.81E-02
Polyethylene	H	7.94E-02
	C	3.97E-02
90-10 Cupronickel*	Cu	7.64E-02
	Ni	8.22E-03
	Mn	8.14E-04
CaSO4	Ca	3.88E-03
	S	6.48E-03
	O	3.14E-02

*(Cubberly, W., 1978:45,374-375)

Appendix C: Sample Input File for MORSE: "Best" Design Case

Below is a copy of the input file for the "best" design case:

```
Problem to determine 14-MeV n-gamma effects in Nova Upgrade Oct 91
 1000 4000     6     1    46    23    46    69     0     0   300     8     0
    4     0     0    01.0         1.000E-5  1.000e+4  1.000e-03 4.384E3
       0.0         0.0         0.0         0.0         0.0         0.0         0.0
 1.9600 +7 1.6900 +7 1.4900 +7 1.4200 +7 1.3800 +7 1.2800 +7 1.2200 +7
 1.1100 +7 1.0000 +7 9.0000 +6 8.2000 +6 7.4000 +6 6.4000 +6 5.0000 +6
 4.7000 +6 4.1000 +6 3.0000 +6 2.4000 +6 2.3000 +6 1.8000 +6 1.4227 +6
 1.1000 +6 9.6164 +5 8.2085 +5 7.4274 +5 6.3928 +5 5.5000 +5 3.6883 +5
 2.4724 +5 1.6000 +5 1.1000 +5 5.2000 +4 3.4307 +4 2.5000 +4 2.1875 +4
 1.0000 +4 3.4000 +3 1.2000 +3 5.8000 +2 2.7536 +2 1.0000 +2 2.9000 +1
 1.1000 +1 3.1000 +0 1.1000 +0 4.1400 -1 2.0     +7 1.4     +7 1.2     +7
 1.0    +7 8.0     +6 7.0     +6 6.0     +6 5.0     +6 4.0     +6 3.0     +6
 2.5    +6 2.0     +6 1.5     +6 1.0     +6 7.0     +5 4.5     +5 3.0     +5
 1.5    +5 1.0     +5 7.0     +4 4.5     +4 3.0     +4 2.0     +4
000045FA231A
    1    1     0     0     0     3    69
    1    1    21     1     1     1    0.75 +0    5.00 -3    1.00
   22    1    31     1     1     1    1.00 +0    1.00 -2    1.00
   32    1    46     1     1     1    2.00 +0    1.00 -2    1.00
   47    1    69     1     1     1    0.75 +0    1.00 -4    4.00
    1    1    69     2     1     2    2.00 +0    1.00 -2    1.00
    1    1    69     3     1     3    2.00 +0    1.00 -2    2.00
   -1
    0    0     0     0
 1.0000 -1 1.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 5.0000 -1 5.0000 -1 5.0000 -1
 5.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
```

```
1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
1.0000 -1 1.0000 -1 1.0000 -1 1.0000 -1
   0     0           COMBINATORIAL GEOMETRY  Nova Upgrade
  SPH    1 00.000+00 00.000+00 00.000+00 00.750+02 00.000+00 00.000+00
  SPH    2  0.0000 0  0.0000 0  0.0000 0  0.50   +2   0.000 0  0.0000 0
  TRC    3  0.0000 0  0.0000 0 -0.250 +2 0.0000   0  0.0000 0 -0.1800+2
            81.250-1  139.75-1
  TRC    4  0.0000 0  0.0000 0 -0.3400+2  0.0000 0  0.0000+0 -0.0900+2
            110.50-1  139.75-1
  SPH    5  0.0000 0  0.0000 0  0.0000 0  4.0000+2  0.0000 0  0.0000 0
  SPH    6  0.0000 0  0.0000 0  0.0000 0  4.0500+2  0.0000 0  0.0000 0
  SPH    7  0.0000 0  0.0000 0  0.0000 0  4.2500+2  0.0000 0  0.0000 0
  SPH    8  0.0000 0  0.0000 0  0.0000 0  5.0000+2  0.0000 0  0.0000 0
  SPH    9  0.0000 0  0.0000 0  0.0000 0  10.000+2  0.0000 0  0.0000 0
  RCC   10  0         0        -3.8000+1 0          0        -3.559 +2
             .7500+2
  RCC   11  0         0        -3.8000+1 0          0        -3.559 +2
            0.7000+2
  RCC   12  0         0        -3.8000+1 0          0        -3.559 +2
            0.6750+2
  RCC   13  0         0        -3.8000+1 0          0        -3.559 +2
            0.6500+2
  RCC   14  0         0        -2.000+ 2 0          0        -0.050 +0
            0.6500+2
  RCC   15  0         0         0.000+ 0 0          0        -3.800 +1
            0.7500+2
  RCC   16  0         0         0.000+ 0 0          0        -3.800 +1
            0.5000+2
  END
  ivd          +5      -1     -10     -15OR    +2     -3OR  +13    -3    -14
          OR  +16      -3
  CON          +1      -2     -10     -15OR   +15    -16
  pgB          +3      -4
  pgL          +4
  lid         +14
  cn1         +10     -11
  cn2         +11     -12
  cn3         +12     -13
  1WL           6      -5
  ply           7      -6
  VOD           9      -7
  END
   3     1     2     2     2     1     1     1     2     3     3     1     1     1
1000     5     6     4     6     5     5     5     5     2     0
46 N, 23 GAMMA (P3) for NOVA Upgrade with Scatterer/Converter Sphere
  46    46    23    23    69    72     4     8    19    31     4     2     1     3
   0     0     0     0     0     0     0   -10     0     0     0
SAMBO ANALYSIS INPUT DATA for Nova Upgrade
   1    43    66   -41     0     6    12     8
```

```
       0.0       00.0     -350.0
RESULTS of Nova Upgrade calculation -- Neutron-Gamma simulation
Gamma Ionization in Silicon {Gy(Si)/source particle (Gammas Dose)}
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        8.5045-12 1.5823-11 2.4318-11
 1.9441-11 2.485 -11 2.175 -11 1.862 -11 1.545 -11 1.222 -11 9.803 -12
 8.284 -12 6.718 -12 5.074 -12 3.681 -12 2.657 -12 1.841 -12 1.120 -12
 7.454 -13 8.359 -13 1.447 -12 3.211 -12 7.3881-12 2.2600-11
Neutron Ionization in Silicon {Gy(Si)/source particle (Neutrons Dose)}
1.4136E-111.1867E-111.1380E-111.1213E-111.1108E-111.0931E-111.0560E-11
9.6552E-128.2573E-127.1892E-126.4233E-123.9406E-121.7265E-121.3041E-12
9.9411E-136.4401E-135.8343E-134.9385E-135.0306E-134.0888E-132.2068E-13
2.2945E-132.6028E-132.5167E-131.4625E-131.9371E-131.2283E-131.0185E-13
1.1136E-135.3753E-157.6044E-154.0843E-152.5145E-151.9299E-151.3058E-15
5.7105E-161.7532E-166.5553E-172.7522E-173.1906E-171.5852E-175.8717E-18
2.5149E-182.7854E-187.5570E-188.8589E-180.0000E+000.0000E+000.0000E+00
0.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+00
0.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+00
0.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+000.0000E+00
Gamma Fluence (gammas/cm2/source particle)
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
Neutron Fluence (neutrons/cm2/source neutron)
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        1.0        1.0        1.0
 1.0        1.0        1.0        1.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
Gammas (joules/eV/source particle)
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
```

```
 0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        1.7e7      1.3e7      1.1e7
 9.0e6      7.5e6      6.5e6      5.5e6      4.5e6      3.5e6      2.75e6
 2.25e6     1.75e6     1.25e6     8.5e5      5.75e5     3.75e5     2.25e5
 1.25e5     8.5e4      5.75e4     3.75e4     2.5e4      1.0e4
Neutrons (joules/eV/source particle)
 1.825e7    1.59e7     1.455e7    1.4e7      1.33e7     1.25e7     1.165e7
 1.055e7    9.5e6      8.6e6      7.8e6      6.9e6      5.7e6      4.85e6
 4.4e6      3.55e6     2.7e6      2.35e6     2.05e6     1.61135e6 1.26135e6
 1.03082e6 8.91245e5 7.81795e5 6.9101e5  5.9363e5   4.59415e5 3.08035e5
 2.0362e5   1.35e5     8.1e4      4.31535e4 2.96535e4 2.34375e4 1.59375e4
 6.7e3      2.3e3      890        427.68     187.68     64.5       20
 7.05       2.1        0.757      0.207      0.0000E+0 0.0000E+0 0.0000E+0
 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0
 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0
 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0 0.0000E+0
{eV/cm2/eV/source neutron}
    4    5    6    7    8    9   10   11   12   13   14   15   16   17
   18   19   20   21   22   23   24   25   26   27   28   29   30   31
   32   33   34   35   36   37   38   39   40   41   42   43   44   45
   46   47   48   49   50   51   52   53   54   55   56   57   58   59
   60   61   62   63   64   65   66   67   68   69
(eV/cm2/sec/Source Particle)
(eV/cm2/eV/sec/source particle)
   1.18e-8    1.2e-8    1.3e-8    1.4e-8    1.5e-8    1.6e-8    1.7e-8
    1.8e-8    1.9e-8    2.0e-8    2.1e-8    2.2e-8    2.3e-8    2.4e-8
    2.5e-8    2.6e-8    2.7e-8    2.8e-8    2.9e-8    3.0e-8    3.1e-8
    3.2e-8    3.3e-8    3.4e-8    3.5e-8    3.6e-8    3.7e-8    3.8e-8
    4.0e-8    5.0e-8    6.0e-8    7.0e-8    8.0e-8    9.0e-8    1.0e-7
    1.0e-6    1.0e-5    1.0e-4    1.0e-3    1.0e-2    1.0e-1
$$$$$$$$$$ Nova Upgrade Neutron-Gamma Effects Simulation 91
$$$$$$$$$$$$$$
```

The following is a key to the media numbers:

1. Glass (SiO_2).
2. Leaded Borated Polyethylene.
3. Copper.
4. 1% Borated Polyethylene.
5. 5083 Aluminum.
6. Tungsten.
7. Polyethylene.
8. Tantalum.

Appendix D: Sample MORSE Output: "Best" Design Case

The following is a portion of the output from the "best" design case with the detector at 350 cm:

```
 TODAY IS  1- 3-92                                 al-conv-al-can at 350m
0Gamma Ionization in Silicon {Gy(Si)/source particle (Gammas Dose)}
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
     DETECTOR    UNCOLL               FSD          TOTAL         FSD
                RESPONSE             UNCOLL       RESPONSE      TOTAL
         1      4.5354E-18           0.03355      8.0955E-18    0.02346
2Neutron Ionization in Silicon {Gy(Si)/source particle (Neutrons Dose)}
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
     DETECTOR       UNCOLL              FSD           TOTAL       FSD
                  RESPONSE             UNCOLL       RESPONSE     TOTAL
         1      1.1729E-19           0.00019        2.4142E-18  0.03607
2Gamma Fluence (gammas/cm2/source particle)
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
     DETECTOR    UNCOLL              FSD            TOTAL         FSD
                RESPONSE            UNCOLL         RESPONSE      TOTAL
        1       4.3667E-07          0.02810       1.7777E-06    0.02268
2Neutron Fluence (neutrons/cm2/source neutron)
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
 DETECTOR           UNCOLL              FSD            TOTAL        FSD
                   RESPONSE            UNCOLL         RESPONSE     TOTAL
    1              1.0461E-08          0.00015        4.9629E-06   0.02739
2Gammas (gammas eV/cm2/sourrce particle)
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
DETECTOR            UNCOLL              FSD            TOTAL        FSD
                   RESPONSE            UNCOLL         RESPONSE     TOTAL
    1              1.3155E+00          0.03434        2.1628E+00   0.02362
2Neutrons (neutrons eV/cm2/source particle)
0RESPONSES(DETECTOR) RESULTS of Nova Upgrade calculation -- Neutron-
Gamma simulation
 DETECTOR           UNCOLL              FSD            TOTAL        FSD
                   RESPONSE            UNCOLL         RESPONSE     TOTAL
    1              1.4645E-01          0.00000       4.5448E+00    0.03176
1             FLUENCE(ENERGY,DETECTOR) {eV/cm2/eV/source neutron}
0   DETECTOR NO.        1
 ENERGIES
   1.960E+07
                  1.485E-14
                     0.071
```

```
    1.380E+07
                  3.981E-14
                     0.053
    1.280E+07
                  8.336E-15
                     0.174
    1.220E+07
                  9.907E-15
                     0.161
    1.110E+07
                  7.766E-15
                     0.136
    1.000E+07
                  6.551E-15
                     0.131
    9.000E+06
                  9.055E-15
                     0.075
    8.200E+06
                  9.135E-15
                     0.155
    7.400E+06
                  1.436E-14
                     0.123
    6.400E+06
                  2.444E-14
                     0.051
    5.000E+06
                  2.606E-14
                     0.121
    4.700E+06
                  3.994E-14
                     0.125
    4.100E+06
                  5.001E-14
                     0.105
    3.000E+06
                  1.346E-13
                     0.077
    2.400E+06
                  1.649E-13
                     0.122
    2.300E+06
                  2.033E-13
                     0.074
    1.800E+06
                  3.022E-13
                     0.033
    1.423E+06
1   FLUENCE(ENERGY,DETECTOR) {eV/cm2/eV/source neutron}
0    DETECTOR NO.        1
 ENERGIES
```

1.423E+06

4.626E-13

0.109

1.100E+06

7.531E-13

0.082

9.616E+05

7.551E-13

0.179

8.209E+05

8.288E-13

0.172

7.427E+05

1.054E-12

0.068

6.393E+05

1.461E-12

0.057

5.500E+05

1.976E-12

0.028

3.688E+05

3.369E-12

0.135

2.472E+05

2.546E-12

0.057

1.600E+05

6.638E-12

0.083

1.100E+05

1.509E-11

0.092

5.200E+04

2.505E-12

0.134

3.431E+04

4.902E 11

0.104

2.500E+04

2.539E-11

0.099

2.188E+04

1.539E-11

0.129

1.000E+04

2.020E-11

0.141

3.400E+03

5.127E 11

0.088

1.200E+03

```
1  FLUENCE(ENERGY,DETECTOR) {eV/cm2/eV/source neutron}
0   DETECTOR NO.        1
 ENERGIES
   1.200E+03
                          1.203E-10
                             0.209
   5.800E+02
                          1.829E-10
                             0.260
   2.754E+02
                          5.005E-10
                             0.179
   1.000E+02
                          1.110E-09
                             0.113
   2.900E+01
                          2.889E-09
                             0.252
   1.100E+01
                          7.695E-09
                             0.236
   3.100E+00
                          1.852E-08
                             0.253
   1.100E+00
                          3.206E-08
                             0.061
   4.140E-01
                          1.216E-08
                             0.123
   1.000E-05
 ENERGIES
   2.000E+07
                          6.376E-18
                             0.693
   1.400E+07
                          1.659E-17
                             1.000
   1.200E+07
                          1.067E-15
                             0.255
   1.000E+07
                          6.349E-15
                             0.280
   8.000E+06
                          1.906E-14
                             0.055
   7.000E+06
                          1.851E-14
                             0.077
   6.000E+06
                          2.999E-14
```

```
                    0.115
  5.000E+06
                 4.157E-14
                    0.102
  4.000E+06
1   FLUENCE(ENERGY,DETECTOR) {eV/cm2/eV/source neutron}
0   DETECTOR NO.        1
 ENERGIES
  4.000E+06
                 8.471E-14
                    0.060
  3.000E+06
                 1.074E-13
                    0.073
  2.500E+06
                 2.162E-13
                    0.063
  2.000E+06
                 1.783E-13
                    0.066
  1.500E+06
                 3.034E-13
                    0.028
  1.000E+06
                 3.938E-13
                    0.063
  7.000E+05
                 6.955E-13
                    0.022
  4.500E+05
                 1.031E-12
                    0.034
  3.000E+05
                 2.345E-12
                    0.047
  1.500E+05
                 4.182E-12
                    0.075
  1.000E+05
                 3.913E-12
                    0.045
  7.000E+04
                 1.559E-12
                    0.083
  4.500E+04
                 1.729E-13
                    0.134
  3.000E+04
                 5.575E-15
                    0.430
  2.000E+04
                 1.282E-18
```

```
                    0.957
   1.000E+04
1DETECTOR NO 1 RESPONSE(RESPONSE,TIME,DETECTOR) (eV/cm2/sec/Source
Particle)
 RESPONSE      1          2          3          4          5          6
TIMES
   1.167E-08
            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000     0.000     0.000     0.000     0.000     0.000
   1.180E-08
            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000     0.000     0.000     0.000     0.000     0.000
   1.200E-08
            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000     0.000     0.000     0.000     0.000     0.000
   1.300E-08
            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000     0.000     0.000     0.000     0.000     0.000
   1.400E-08
            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00
            0.000     0.000     0.000     0.000     0.000     0.000
   1.500E-08
             1.050E-15 0.000E+00 2.069E-04 0.000E+00 2.586E+02 0.000E+00
             0.000     0.000     1.000     0.000     1.000     0.000
   1.600E-08
             9.319E-12 0.000E+00 1.675E+00 0.000E+00 2.380E+06 0.000E+00
             0.000     0.000     0.614     0.000     0.623     0.000
   1.700E-08
             4.278E-11 0.000E+00 4.277E+00 0.000E+00 1.208E+07 0.000E+00
             0.000     0.000     0.764     0.000     0.925     0.000
   1.800E-08
             2.454E-12 0.000E+00 3.413E-01 0.000E+00 6.633E+05 0.000E+00
             0.000     0.000     0.585     0.000     0.620     0.000
   1.900E-08
             1.172E-12 0.000E+00 4.828E-01 0.000E+00 2.555E+05 0.000E+00
             0.000     0.000     0.578     0.000     0.776     0.000
   2.000E-08
             6.764E-12 0.000E+00 5.130E+00 0.000E+00 1.154E+06 0.000E+00
             0.000     0.000     0.980     0.000     0.964     0.000
   2.100E-08
             3.665E-10 0.000E+00 3.927E+01 0.000E+00 1.068E+08 0.000E+00
             0.000     0.000     0.090     0.000     0.100     0.000
   2.200E-08
             9.090E-10 0.000E+00 1.053E+02 0.000E+00 2.588E+08 0.000E+00
             0.000     0.000     0.031     0.000     0.049     0.000
   2.300E-08
             9.510E-10 0.000E+00 1.411E+02 0.000E+00 2.631E+08 0.000E+00
             0.000     0.000     0.033     0.000     0.039     0.000
   2.400E-08
             6.066E-10 0.000E+00 1.191E+02 0.000E+00 1.626E+08 0.000E+00
             0.000     0.000     0.108     0.000     0.058     0.000
   2.500E-08
```

```
            4.149E-10 0.000E+00 9.185E+01 0.000E+00 1.107E+08 0.000E+00
            0.000     0.000     0.122     0.000     0.141     0.000
   2.600E-08
            3.321E-10 0.000E+00 8.905E+01 0.000E+00 8.840E+07 0.000E+00
            0.000     0.000     0.115     0.000     0.040     0.000
   2.700E-08
1DETECTOR NO  1     RESPONSE(RESPONSE,TIME,DETECTOR) (eV/cm2/sec/Source
Particle)
RESPONSE          1         2         3         4         5         6
TIMES
   2.700E-08
            2.407E-10 0.000E+00 6.025E+01 0.000E+00 6.266E+07 0.000E+00
            0.000     0.000     0.154     0.000     0.051     0.000
   2.800E-08
            2.253E-10 0.000E+00 6.858E+01 0.000E+00 5.824E+07 0.000E+00
            0.000     0.000     0.143     0.000     0.048     0.000
   2.900E-08
            1.765E-10 0.000E+00 5.034E+01 0.000E+00 4.547E+07 0.000E+00
            0.000     0.000     0.168     0.000     0.139     0.000
   3.000E-08
            1.127E-10 0.000E+00 2.750E+01 0.000E+00 3.033E+07 0.000E+00
            0.000     0.000     0.118     0.000     0.147     0.000
   3.100E-08
            1.516E-10 0.000E+00 4.263E+01 0.000E+00 3.971E+07 0.000E+00
            0.000     0.000     0.261     0.000     0.074     0.000
   3.200E-08
            8.657E-11 0.000E+00 2.693E+01 0.000E+00 2.177E+07 0.000E+00
            0.000     0.000     0.147     0.000     0.187     0.000
   3.300E-08
            9.198E-11 0.000E+00 2.304E+01 0.000E+00 2.447E+07 0.000E+00
            0.000     0.000     0.247     0.000     0.211     0.000
   3.400E-08
            7.685E-11 0.000E+00 1.984E+01 0.000E+00 1.941E+07 0.000E+00
            0.000     0.000     0.192     0.000     0.185     0.000
   3.500E-08
            7.601E-11 0.000E+00 1.837E+01 0.000E+00 1.940E+07 0.000E+00
            0.000     0.000     0.051     0.000     0.133     0.000
   3.600E-08
            7.700E-11 0.000E+00 1.861E+01 0.000E+00 2.030E+07 0.000E+00
            0.000     0.000     0.109     0.000     0.196     0.000
   3.700E-08
            6.510E-11 0.000E+00 1.925E+01 0.000E+00 1.664E+07 0.000E+00
            0.000     0.000     0.290     0.000     0.205     0.000
   3.800E-08
            8.696E-11 0.000E+00 1.929E+01 0.000E+00 2.353E+07 0.000E+00
            0.000     0.000     0.195     0.000     0.093     0.000
   4.000E-08
            6.189E-11 0.000E+00 1.431E+01 0.000E+00 1.648E+07 0.000E+00
            0.000     0.000     0.144     0.000     0.081     0.000
   5.000E-08
            1.119E-11 0.000E+00 4.430E+00 0.000E+00 2.570E+06 0.000E+00
            0.000     0.000     0.199     0.000     0.157     0.000
```

```
    6.000E-08
                 1.503E-11 2.975E-11 3.478E+00 2.656E+00 3.911E+06 3.699E+07
                 0.000     0.000     0.192     0.122     0.257     0.122
    7.000E-08
                 1.096E-11 9.103E-11 2.521E+00 8.208E+00 2.815E+06 1.109E+08
                 0.000     0.000     0.288     0.075     0.195     0.077
    8.000E-08
                 3.287E-11 3.611E-11 9.858E+00 3.587E+00 8.209E+06 4.345E+07
                 0.000     0.000     0.505     0.019     0.242     0.016
    9.000E-08
1DETECTOR NO  1     RESPONSE(RESPONSE,TIME,DETECTOR) (eV/cm2/sec/Source
Particle)
  RESPONSE            1         2         3         4         5         6
 TIMES
    9.000E-08
                 2.693E-11 1.318E-11 6.178E+00 2.093E+00 8.165E+06 1.782E+07
                 0.404     0.000     0.330     0.054     0.468     0.055
    1.000E-07
                 9.427E-13 7.600E-13 2.872E-01 2.395E+00 2.293E+05 2.539E+06
                 0.000     0.000     0.084     0.040     0.118     0.030
    1.000E-06
                 2.450E-14 3.290E-15 4.899E-03 2.438E-01 6.552E+03 1.851E+04
                 0.000     0.000     0.114     0.046     0.068     0.059
    1.000E-05
                      1.975E-15 1.361E-18 4.335E-C* 3.938E-03 5.228E+02
1.480E+01
                         0.000     0.000     0.290     0.080     0.222
0.570
    1.000E-04
                 7.218E-17 4.940E-22 1.698E-05 9.744E-05 1.881E+01 9.099E-04
                 0.000     0.000     0.128     0.218     0.133     0.363
    1.000E-03
                 0.000E+00 4.161E-25 0.000E+00 8.090E-08 0.000E+00 1.162E-07
                 0.000     0.000     0.000     0.521     0.000     0.704
    1.000E-02
                 0.000E+00 2.689E-25 0.000E+00 3.035E-08 0.000E+00 6.282E-09
                 0.000     0.000     0.000     0.123     0.000     0.123
    1.758E-01
0TIME REQUIRED FOR THE PRECEDING   6 BATCHES WAS 31 SECOND(S), 55
MINUTE(S),   1 HOUR(S)
1NEUTRON DEATHS                        NUMBER               WEIGHT
0KILLED BY RUSSIAN ROULETTE           153177            3.05511E+02
 ESCAPED                                1504            6.75745E+02
 REACHED ENERGY CUTOFF                     0            0.00000E+00
 REACHED TIME CUTOFF                     537            5.51033E+01
0
0NUMBER OF SCATTERINGS
0MEDIUM            NUMBER
    1                 0
    2            350460
    3                 0
    4              3213
```

5	1356602
6	7880
7	0
8	0
TOTAL	1718155

1REAL SCATTERING COUNTERS

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00
4	7788	6.39E+03	782	6.17E+02	1119	7.66E+02
5	2708	1.11E+03	281	1.48E+02	729	3.80E+02
6	235	9.40E+01	28	1.09E+01	50	2.40E+01
7	650	2.51E+02	70	3.15E+01	188	1.04E+02
8	552	2.16E+02	62	2.80E+01	184	9.19E+01
9	312	1.21E+02	29	1.62E+01	143	7.39E+01
10	348	1.36E+02	53	2.54E+01	155	8.64E+01
11	520	2.09E+02	73	3.14E+01	173	9.45E+01
12	853	3.40E+02	123	5.34E+01	282	1.37E+02
13	2198	8.89E+02	267	1.27E+02	627	3.02E+02
14	559	2.24E+02	67	3.65E+01	152	7.25E+01
15	1574	6.27E+02	191	8.52E+01	437	1.94E+02
16	3961	1.60E+03	537	2.37E+02	1196	5.66E+02
17	5232	2.12E+03	815	3.63E+02	1287	6.05E+02
18	820	3.33E+02	129	5.42E+01	221	1.03E+02
19	6714	2.73E+03	1114	5.05E+02	1564	7.31E+02
20	7351	2.97E+03	1316	5.85E+02	1837	8.57E+02
21	8548	3.48E+03	1574	6.99E+02	2081	9.51E+02
22	5255	2.23E+03	1071	4.87E+02	1456	6.65E+02
23	5616	2.41E+03	1233	5.63E+02	1514	6.85E+02
24	4512	1.98E+03	1066	4.91E+02	1073	4.86E+02
25	5747	2.58E+03	1311	6.04E+02	1571	7.16E+02
26	6147	2.77E+03	1395	6.49E+02	1610	7.37E+02
27	17167	7.69E+03	3782	1.71E+03	4449	2.04E+03
28	17193	7.77E+03	4526	2.09E+03	5082	2.35E+03
29	9221	4.28E+03	2603	1.23E+03	4932	2.28E+03
30	14387	6.75E+03	3658	1.71E+03	3940	1.82E+03
31	36822	1.71E+04	9349	4.26E+03	9145	4.18E+03
32	3078	1.45E+03	1167	5.49E+02	4515	2.09E+03
33	18159	8.39E+03	4543	2.05E+03	4555	2.08E+03
34	940	4.44E+02	306	1.46E+02	1874	8.62E+02
35	1992	9.28E+02	1113	5.20E+02	9861	4.46E+03
36	1267	5.37E+02	1457	6.68E+02	14231	6.43E+03
37	761	3.16E+02	1592	7.20E+02	12630	5.70E+03
38	319	1.21E+02	898	3.82E+02	9167	4.09E+03
39	522	2.10E+02	1609	6.97E+02	9263	4.08E+03
40	265	1.07E+02	1030	4.20E+02	12352	5.32E+03
41	273	1.03E+02	987	4.16E+02	15269	6.28E+03
42	184	6.25E+01	702	2.78E+02	12474	4.77E+03
43	175	5.91E+01	927	3.23E+02	15234	5.17E+03
44	187	5.03E+01	671	1.88E+02	13157	3.66E+03

45	164	4.16E+01	701	1.53E+02	16938	3.33E+03
46	5	1.96E+00	709	7.80E+01	50395	4.19E+03
47	4	2.13E+00	0	0.00E+00	0	0.00E+00
48	2	8.08E-01	1	4.16E-01	1	3.92E-01
49	91	4.09E+01	4	2.97E+00	14	8.96E+00
50	3022	1.58E+02	237	1.24E+01	550	3.11E+01
51	7611	2.22E+02	767	4.26E+01	1532	7.38E+01
52	3178	2.89E+02	293	4.83E+01	623	8.81E+01
53	4817	4.87E+02	409	4.58E+01	792	1.01E+02
54	13153	6.67E+02	1128	8.60E+01	1918	2.42E+02
55	18591	1.53E+03	1458	1.95E+02	2288	2.89E+02
56	8403	9.02E+02	743	1.63E+02	1115	3.07E+02
57	12263	1.99E+03	1005	2.50E+02	1555	4.45E+02
58	12764	2.03E+03	1055	3.07E+02	1531	5.46E+02
59	26922	3.55E+03	2374	7.08E+02	2635	7.65E+02
60	30433	3.39E+03	2524	6.19E+02	3408	9.47E+02
61	52592	5.39E+03	6495	1.54E+03	30277	7.19E+03
62	54529	6.05E+03	5919	1.44E+03	15662	2.52E+03
63	164910	1.81E+04	15652	3.53E+03	28603	3.50E+03
64	146920	1.46E+04	13080	2.77E+03	6751	5.31E+02
65	167542	1.21E+04	13217	2.07E+03	1511	1.79E+02
66	196399	6.30E+03	12839	1.14E+03	550	3.96E+01
67	93132	8.33E+02	2949	1.48E+02	30	2.50E+00
68	10666	2.62E+01	138	6.72E+00	2	3.46E+00
69	265	1.58E-01	1	8.91E-01	0	0.00E+00

1SECONDARY PRODUCTION COUNTERS (BOTH THE GROUPS CAUSING PRODUCTION AND RESULTING FROM PRODUCTION

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00
4	3922	3.23E+03	76	6.34E+01	129	8.70E+01
5	1365	5.57E+02	33	2.12E+01	87	4.57E+01
6	121	4.87E+01	3	1.21E+00	3	1.16E+00
7	324	1.24E+02	5	2.37E+00	21	1.25E+01
8	271	1.07E+02	2	8.31E-01	18	1.14E+01
9	144	5.46E+01	2	1.20E+00	14	7.43E+00
10	167	6.45E+01	5	1.81E+00	20	1.31E+01
11	256	1.03E+02	4	1.86E+00	24	1.21E+01
12	466	1.85E+02	16	7.17E+00	26	1.12E+01
13	1105	4.46E+02	23	9.95E+00	70	3.24E+01
14	284	1.14E+02	6	4.26E+00	14	5.22E+00
15	777	3.09E+02	23	1.09E+01	50	2.24E+01
16	1986	8.04E+02	54	2.33E+01	114	5.67E+01
17	2593	1.05E+03	83	3.49E+01	118	5.40E+01
18	435	1.76E+02	13	4.84E+00	20	8.59E+00
19	3362	1.37E+03	114	4.94E+01	175	8.34E+01
20	3667	1.48E+03	137	6.04E+01	191	8.39E+01
21	4307	1.76E+03	153	7.10E+01	227	1.04E+02
22	2597	1.11E+03	116	5.20E+01	118	5.81E+01
23	2795	1.21E+03	150	6.55E+01	153	7.05E+01

24	2252	9.80E+02	105	4.68E+01	109	4.84E+01
25	2908	1.30E+03	130	6.15E+01	140	6.74E+01
26	3107	1.41E+03	137	6.40E+01	168	7.68E+01
27	8487	3.80E+03	392	1.82E+02	427	1.97E+02
28	8658	3.91E+03	423	1.99E+02	557	2.65E+02
29	4703	2.17E+03	244	1.13E+02	495	2.32E+02
30	7112	3.33E+03	353	1.64E+02	413	1.93E+02
31	18545	8.62E+03	952	4.38E+02	907	4.13E+02
32	1570	7.42E+02	108	5.09E+01	468	2.13E+02
33	9172	4.25E+03	445	2.01E+02	431	1.96E+02
34	521	2.44E+02	33	1.47E+01	181	8.03E+01
35	990	4.63E+02	107	4.99E+01	1008	4.59E+02
36	622	2.69E+02	136	5.99E+01	1443	6.64E+02
37	393	1.61E+02	132	6.08E+01	1299	5.78E+02
38	173	6.64E+01	79	3.52E+01	924	4.06E+02
39	274	1.11E+02	160	6.83E+01	984	4.40E+02
40	128	5.18E+01	93	3.86E+01	1233	5.32E+02
41	145	5.53E+01	92	4.10E+01	1487	6.06E+02
42	93	3.02E+01	62	2.52E+01	1242	4.81E+02
43	79	2.60E+01	85	2.88E+01	1529	5.25E+02
44	98	2.53E+01	68	1.77E+01	1378	3.83E+02
45	81	2.03E+01	60	1.44E+01	1758	3.49E+02
46	1	4.04E-01	74	8.81E+00	5122	4.29E+02
47	3	2.13E+00	0	0.00E+00	0	0.00E+00
48	2	1.62E+00	0	0.00E+00	0	0.00E+00
49	37	4.47E+01	1	3.38E+00	1	5.31E+00
50	3472	1.80E+02	179	1.29E+01	4	2.80E-02
51	9022	2.65E+02	597	4.88E+01	122	9.79E+00
52	3040	3.28E+02	174	2.89E+01	94	3.37E+01
53	4363	4.99E+02	297	2.91E+01	27	2.82E+01
54	13257	6.65E+02	713	5.34E+01	103	1.26E+02
55	16160	1.47E+03	840	1.32E+02	87	6.46E+01
56	5456	7.29E+02	291	9.78E+01	156	1.50E+02
57	6890	1.66E+03	323	1.61E+02	288	1.36E+02
58	4608	1.29E+03	224	2.18E+02	176	1.45E+02
59	12557	1.84E+03	455	3.79E+02	115	1.13E+02
60	11004	1.37E+03	460	2.38E+02	700	2.51E+02
61	1214	4.61E+01	311	1.01E+02	23234	4.57E+03
62	865	2.78E+01	64	1.51E+02	171	8.08E+01
63	375	5.93E+01	39	1.13E+02	10	3.66E-01
64	0	0.00E+00	49	1.94E+02	1	1.47E-02
65	0	0.00E+00	4	7.92E+00	5	2.14E+00
66	1	8.44E-01	24	7.54E+01	0	0.00E+00
67	8650	1.05E+01	439	6.17E+00	0	0.00E+00
68	0	0.00E+00	3	1.02E+00	1	3.46E+00
69	0	0.00E+00	1	8.91E-01	0	0.00E+00

1NUMBER OF SPLITTINGS

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00

4	1320	5.85E+02	8	1.02E+01	0	0.00E+00
5	875	3.88E+02	4	4.62E+00	0	0.00E+00
6	60	2.66E+01	0	0.00E+00	0	0.00E+00
7	143	6.33E+01	0	0.00E+00	0	0.00E+00
8	130	5.75E+01	0	0.00E+00	0	0.00E+00
9	54	2.39E+01	0	0.00E+00	0	0.00E+00
10	79	3.53E+01	0	0.00E+00	0	0.00E+00
11	122	5.46E+01	0	0.00E+00	0	0.00E+00
12	159	7.04E+01	0	0.00E+00	0	0.00E+00
13	343	1.52E+02	0	0.00E+00	0	0.00E+00
14	62	2.74E+01	0	0.00E+00	0	0.00E+00
15	139	6.21E+01	0	0.00E+00	0	0.00E+00
16	337	1.50E+02	0	0.00E+00	0	0.00E+00
17	267	1.19E+02	0	0.00E+00	0	0.00E+00
18	40	1.77E+01	0	0.00E+00	0	0.00E+00
19	267	1.20E+02	3	3.84E+00	0	0.00E+00
20	226	1.02E+02	0	0.00E+00	0	0.00E+00
21	205	9.42E+01	1	1.20E+00	0	0.00E+00
22	3	2.26E+00	0	0.00E+00	0	0.00E+00
23	6	3.93E+00	1	1.20E+00	0	0.00E+00
24	3	2.20E+00	1	1.28E+00	0	0.00E+00
25	15	1.08E+01	0	0.00E+00	0	0.00E+00
26	9	6.29E+00	0	0.00E+00	0	0.00E+00
27	20	1.46E+01	2	2.09E+00	0	0.00E+00
28	12	8.12E+00	0	0.00E+00	0	0.00E+00
29	5	3.59E+00	0	0.00E+00	0	0.00E+00
30	6	4.49E+00	0	0.00E+00	0	0.00E+00
31	5	2.97E+00	0	0.00E+00	0	0.00E+00
32	0	0.00E+00	0	0.00E+00	0	0.00E+00
33	0	0.00E+00	0	0.00E+00	0	0.00E+00
34	0	0.00E+00	0	0.00E+00	0	0.00E+00
35	0	0.00E+00	0	0.00E+00	0	0.00E+00
36	0	0.00E+00	0	0.00E+00	0	0.00E+00
37	0	0.00E+00	0	0.00E+00	0	0.00E+00
38	0	0.00E+00	0	0.00E+00	0	0.00E+00
39	0	0.00E+00	0	0.00E+00	0	0.00E+00
40	0	0.00E+00	0	0.00E+00	0	0.00E+00
41	0	0.00E+00	0	0.00E+00	0	0.00E+00
42	0	0.00E+00	0	0.00E+00	0	0.00E+00
43	0	0.00E+00	0	0.00E+00	0	0.00E+00
44	0	0.00E+00	0	0.00E+00	0	0.00E+00
45	0	0.00E+00	0	0.00E+00	0	0.00E+00
46	0	0.00E+00	0	0.00E+00	0	0.00E+00
47	1	3.81E-01	0	0.00E+00	0	0.00E+00
48	2	8.08E-01	0	0.00E+00	0	0.00E+00
49	66	3.62E+01	1	1.69E+00	3	5.31E+00
50	243	1.31E+02	8	1.48E+01	0	0.00E+00
51	359	1.91E+02	22	4.03E+01	1	1.18E+00
52	428	2.28E+02	14	2.37E+01	15	2.56E+01
53	648	3.46E+02	8	1.25E+01	17	3.00E+01
54	872	4.65E+02	20	3.49E+01	56	9.19E+01
55	1754	9.40E+02	64	1.42E+02	26	4.11E+01

56	799	4.27E+02	53	1.18E+02	50	7.58E+01
57	1730	9.22E+02	55	1.21E+02	41	6.14E+01
58	1636	8.80E+02	121	2.66E+02	51	7.85E+01
59	1139	6.00E+02	137	3.49E+02	48	8.16E+01
60	731	3.88E+02	81	1.65E+02	77	1.14E+02
61	226	1.08E+02	40	9.70E+01	244	3.36E+02
62	77	3.76E+01	116	3.09E+02	37	5.64E+01
63	101	5.35E+01	67	1.60E+02	3	3.46E+00
64	11	5.67E+00	96	2.37E+02	0	0.00E+00
65	5	2.53E+00	3	6.44E+00	1	1.04E+00
66	3	1.31E+00	38	9.80E+01	0	0.00E+00
67	1	4.96E-01	0	0.00E+00	0	0.00E+00
68	0	0.00E+00	0	0.00E+00	1	1.73E+00
69	0	0.00E+00	0	0.00E+00	0	0.00E+00

1NUMBER OF SPLITTINGS PREVENTED BY LACK OF ROOM

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00
4	0	0.00E+00	0	0.00E+00	0	0.00E+00
5	0	0.00E+00	0	0.00E+00	0	0.00E+00
6	0	0.00E+00	0	0.00E+00	0	0.00E+00
7	0	0.00E+00	0	0.00E+00	0	0.00E+00
8	0	0.00E+00	0	0.00E+00	0	0.00E+00
9	0	0.00E+00	0	0.00E+00	0	0.00E+00
10	0	0.00E+00	0	0.00E+00	0	0.00E+00
11	0	0.00E+00	0	0.00E+00	0	0.00E+00
12	0	0.00E+00	0	0.00E+00	0	0.00E+00
13	0	0.00E+00	0	0.00E+00	0	0.00E+00
14	0	0.00E+00	0	0.00E+00	0	0.00E+00
15	0	0.00E+00	0	0.00E+00	0	0.00E+00
16	0	0.00E+00	0	0.00E+00	0	0.00E+00
17	0	0.00E+00	0	0.00E+00	0	0.00E+00
18	0	0.00E+00	0	0.00E+00	0	0.00E+00
19	0	0.00E+00	0	0.00E+00	0	0.00E+00
20	0	0.00E+00	0	0.00E+00	0	0.00E+00
21	0	0.00E+00	0	0.00E+00	0	0.00E+00
22	0	0.00E+00	0	0.00E+00	0	0.00E+00
23	0	0.00E+00	0	0.00E+00	0	0.00E+00
24	0	0.00E+00	0	0.00E+00	0	0.00E+00
25	0	0.00E+00	0	0.00E+00	0	0.00E+00
26	0	0.00E+00	0	0.00E+00	0	0.00E+00
27	0	0.00E+00	0	0.00E+00	0	0.00E+00
28	0	0.00E+00	0	0.00E+00	0	0.00E+00
29	0	0.00E+00	0	0.00E+00	0	0.00E+00
30	0	0.00E+00	0	0.00E+00	0	0.00E+00
31	0	0.00E+00	0	0.00E+00	0	0.00E+00
32	0	0.00E+00	0	0.00E+00	0	0.00E+00
33	0	0.00E+00	0	0.00E+00	0	0.00E+00
34	0	0.00E+00	0	0.00E+00	0	0.00E+00
35	0	0.00E+00	0	0.00E+00	0	0.00E+00

36	0	0.00E+00	0	0.00E+00	0	0.00E+00
37	0	0.00E+00	0	0.00E+00	0	0.00E+00
38	0	0.00E+00	0	0.00E+00	0	0.00E+00
39	0	0.00E+00	0	0.00E+00	0	0.00E+00
40	0	0.00E+00	0	0.00E+00	0	0.00E+00
41	0	0.00E+00	0	0.00E+00	0	0.00E+00
42	0	0.00E+00	0	0.00E+00	0	0.00E+00
43	0	0.00E+00	0	0.00E+00	0	0.00E+00
44	0	0.00E+00	0	0.00E+00	0	0.00E+00
45	0	0.00E+00	0	0.00E+00	0	0.00E+00
46	0	0.00E+00	0	0.00E+00	0	0.00E+00
47	0	0.00E+00	0	0.00E+00	0	0.00E+00
48	0	0.00E+00	0	0.00E+00	0	0.00E+00
49	0	0.00E+00	0	0.00E+00	0	0.00E+00
50	0	0.00E+00	0	0.00E+00	0	0.00E+00
51	0	0.00E+00	0	0.00E+00	0	0.00E+00
52	0	0.00E+00	0	0.00E+00	0	0.00E+00
53	0	0.00E+00	0	0.00E+00	0	0.00E+00
54	0	0.00E+00	0	0.00E+00	0	0.00E+00
55	0	0.00E+00	0	0.00E+00	0	0.00E+00
56	0	0.00E+00	0	0.00E+00	0	0.00E+00
57	0	0.00E+00	0	0.00E+00	0	0.00E+00
58	0	0.00E+00	0	0.00E+00	0	0.00E+00
59	0	0.00E+00	0	0.00E+00	0	0.00E+00
60	0	0.00E+00	0	0.00E+00	0	0.00E+00
61	0	0.00E+00	0	0.00E+00	0	0.00E+00
62	0	0.00E+00	0	0.00E+00	0	0.00E+00
63	0	0.00E+00	0	0.00E+00	0	0.00E+00
64	0	0.00E+00	0	0.00E+00	0	0.00E+00
65	0	0.00E+00	0	0.00E+00	0	0.00E+00
66	0	0.00E+00	0	0.00E+00	0	0.00E+00
67	0	0.00E+00	0	0.00E+00	0	0.00E+00
68	0	0.00E+00	0	0.00E+00	0	0.00E+00
69	0	0.00E+00	0	0.00E+00	0	0.00E+00

1NUMBER OF RUSSIAN ROULETTE KILLS

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00
4	0	0.00E+00	0	0.00E+00	0	0.00E+00
5	0	0.00E+00	0	0.00E+00	0	0.00E+00
6	0	0.00E+00	0	0.00E+00	0	0.00E+00
7	0	0.00E+00	0	0.00E+00	0	0.00E+00
8	0	0.00E+00	0	0.00E+00	0	0.00E+00
9	0	0.00E+00	0	0.00E+00	0	0.00E+00
10	0	0.00E+00	0	0.00E+00	0	0.00E+00
11	0	0.00E+00	0	0.00E+00	0	0.00E+00
12	0	0.00E+00	0	0.00E+00	0	0.00E+00
13	0	0.00E+00	0	0.00E+00	0	0.00E+00
14	0	0.00E+00	0	0.00E+00	0	0.00E+00
15	0	0.00E+00	0	0.00E+00	0	0.00E+00

16	0	0.00E+00	0	0.00E+00	0	0.00E+00
17	0	0.00E+00	0	0.00E+00	0	0.00E+00
18	0	0.00E+00	0	0.00E+00	0	0.00E+00
19	0	0.00E+00	0	0.00E+00	0	0.00E+00
20	0	0.00E+00	0	0.00E+00	0	0.00E+00
21	0	0.00E+00	0	0.00E+00	0	0.00E+00
22	0	0.00E+00	0	0.00E+00	0	0.00E+00
23	0	0.00E+00	0	0.00E+00	0	0.00E+00
24	0	0.00E+00	0	0.00E+00	0	0.00E+00
25	0	0.00E+00	0	0.00E+00	0	0.00E+00
26	0	0.00E+00	0	0.00E+00	0	0.00E+00
27	0	0.00E+00	0	0.00E+00	0	0.00E+00
28	0	0.00E+00	0	0.00E+00	0	0.00E+00
29	0	0.00E+00	0	0.00E+00	0	0.00E+00
30	0	0.00E+00	0	0.00E+00	0	0.00E+00
31	0	0.00E+00	0	0.00E+00	0	0.00E+00
32	0	0.00E+00	0	0.00E+00	0	0.00E+00
33	0	0.00E+00	0	0.00E+00	0	0.00E+00
34	0	0.00E+00	0	0.00E+00	0	0.00E+00
35	0	0.00E+00	0	0.00E+00	0	0.00E+00
36	0	0.00E+00	0	0.00E+00	0	0.00E+00
37	0	0.00E+00	0	0.00E+00	0	0.00E+00
38	0	0.00E+00	1	9.70E-03	0	0.00E+00
39	0	0.00E+00	1	9.10E-03	0	0.00E+00
40	0	0.00E+00	1	8.89E-03	0	0.00E+00
41	0	0.00E+00	0	0.00E+00	0	0.00E+00
42	0	0.00E+00	0	0.00E+00	0	0.00E+00
43	0	0.00E+00	2	4.76E-03	0	0.00E+00
44	0	0.00E+00	0	0.00E+00	1	9.90E-03
45	0	0.00E+00	1	5.97E-03	11	9.89E-02
46	0	0.00E+00	85	6.75E-01	9371	7.11E+01
47	0	0.00E+00	0	0.00E+00	0	0.00E+00
48	0	0.00E+00	0	0.00E+00	0	0.00E+00
49	0	0.00E+00	0	0.00E+00	0	0.00E+00
50	2	1.81E-04	172	2.84E-01	7	1.66E-02
51	1	9.38E-05	352	1.14E+00	86	4.19E-01
52	1	6.13E-05	138	3.79E-01	87	4.09E-01
53	2	1.62E-04	209	5.06E-01	69	1.69E-01
54	1	8.21E-05	604	1.60E+00	142	3.78E-01
55	3	2.77E-04	811	1.86E+00	264	4.86E-01
56	1	7.80E-05	364	7.10E-01	256	5.40E-01
57	0	0.00E+00	436	8.01E-01	349	6.71E-01
58	2	1.77E-04	518	8.22E-01	463	8.26E-01
59	0	0.00E+00	729	9.13E-01	699	9.78E-01
60	1	7.80E-05	907	1.36E+00	760	1.35E+00
61	1	8.87E-05	2421	3.30E+00	9874	3.63E+01
62	0	0.00E+00	1824	2.43E+00	3110	1.51E+01
63	0	0.00E+00	4117	6.50E+00	14379	6.39E+01
64	12	1.15E-03	2830	4.82E+00	10686	3.57E+01
65	2082	1.88E-01	2628	4.82E+00	3698	6.91E+00
66	4512	1.84E+00	3714	1.75E+01	921	1.85E+00
67	33672	1.87E+00	2701	1.20E+01	114	1.84E-01

68	10740	4.23E-01	357	1.30E+00	9	1.53E-02
69	851	2.17E-02	14	2.98E-02	0	0.00E+00

1NUMBER OF RUSSIAN ROULETTE SURVIVALS

0ENERGY	REGION 1		REGION 2		REGION 3	
GROUP	NUMBER	WEIGHT	NUMBER	WEIGHT	NUMBER	WEIGHT
1	0	0.00E+00	0	0.00E+00	0	0.00E+00
2	0	0.00E+00	0	0.00E+00	0	0.00E+00
3	0	0.00E+00	0	0.00E+00	0	0.00E+00
4	0	0.00E+00	0	0.00E+00	0	0.00E+00
5	0	0.00E+00	0	0.00E+00	0	0.00E+00
6	0	0.00E+00	0	0.00E+00	0	0.00E+00
7	0	0.00E+00	0	0.00E+00	0	0.00E+00
8	0	0.00E+00	0	0.00E+00	0	0.00E+00
9	0	0.00E+00	0	0.00E+00	0	0.00E+00
10	0	0.00E+00	0	0.00E+00	0	0.00E+00
11	0	0.00E+00	0	0.00E+00	0	0.00E+00
12	0	0.00E+00	0	0.00E+00	0	0.00E+00
13	0	0.00E+00	0	0.00E+00	0	0.00E+00
14	0	0.00E+00	0	0.00E+00	0	0.00E+00
15	0	0.00E+00	0	0.00E+00	0	0.00E+00
16	0	0.00E+00	0	0.00E+00	0	0.00E+00
17	0	0.00E+00	0	0.00E+00	0	0.00E+00
18	0	0.00E+00	0	0.00E+00	0	0.00E+00
19	0	0.00E+00	0	0.00E+00	0	0.00E+00
20	0	0.00E+00	0	0.00E+00	0	0.00E+00
21	0	0.00E+00	0	0.00E+00	0	0.00E+00
22	0	0.00E+00	0	0.00E+00	0	0.00E+00
23	0	0.00E+00	0	0.00E+00	0	0.00E+00
24	0	0.00E+00	0	0.00E+00	0	0.00E+00
25	0	0.00E+00	0	0.00E+00	0	0.00E+00
26	0	0.00E+00	0	0.00E+00	0	0.00E+00
27	0	0.00E+00	0	0.00E+00	0	0.00E+00
28	0	0.00E+00	0	0.00E+00	0	0.00E+00
29	0	0.00E+00	0	0.00E+00	0	0.00E+00
30	0	0.00E+00	0	0.00E+00	0	0.00E+00
31	0	0.00E+00	0	0.00E+00	0	0.00E+00
32	0	0.00E+00	0	0.00E+00	0	0.00E+00
33	0	0.00E+00	0	0.00E+00	0	0.00E+00
34	0	0.00E+00	0	0.00E+00	0	0.00E+00
35	0	0.00E+00	0	0.00E+00	0	0.00E+00
36	0	0.00E+00	0	0.00E+00	0	0.00E+00
37	0	0.00E+00	0	0.00E+00	0	0.00E+00
38	0	0.00E+00	0	0.00E+00	0	0.00E+00
39	0	0.00E+00	0	0.00E+00	0	0.00E+00
40	0	0.00E+00	0	0.00E+00	0	0.00E+00
41	0	0.00E+00	0	0.00E+00	0	0.00E+00
42	0	0.00E+00	0	0.00E+00	0	0.00E+00
43	0	0.00E+00	0	0.00E+00	0	0.00E+00
44	0	0.00E+00	0	0.00E+00	0	0.00E+00
45	0	0.00E+00	0	0.00E+00	0	0.00E+00
46	0	0.00E+00	0	0.00E+00	26	1.89E-01
47	0	0.00E+00	0	0.00E+00	0	0.00E+00

48	0	0.00E+00	0	0.00E+00	0	0.00E+00
49	0	0.00E+00	0	0.00E+00	0	0.00E+00
50	0	0.00E+00	1	3.15E-03	0	0.00E+00
51	0	0.00E+00	0	0.00E+00	1	5.28E-03
52	0	0.00E+00	1	2.98E-03	1	4.30E-03
53	0	0.00E+00	0	0.00E+00	0	0.00E+00
54	0	0.00E+00	0	0.00E+00	1	3.74E-04
55	0	0.00E+00	1	2.99E-03	0	0.00E+00
56	0	0.00E+00	1	2.51E-03	0	0.00E+00
57	0	0.00E+00	2	4.04E-03	0	0.00E+00
58	0	0.00E+00	0	0.00E+00	0	0.00E+00
59	0	0.00E+00	0	0.00E+00	0	0.00E+00
60	0	0.00E+00	3	5.97E-03	1	6.36E-03
61	0	0.00E+00	4	1.00E-02	13	8.56E-02
62	0	0.00E+00	1	8.30E-03	12	8.71E-02
63	0	0.00E+00	6	2.64E-02	37	2.11E-01
64	0	0.00E+00	7	3.84E-02	18	8.54E-02
65	0	0.00E+00	1	2.83E-03	2	2.80E-03
66	0	0.00E+00	16	1.09E-01	0	0.00E+00
67	1	3.09E-05	12	8.83E-02	0	0.00E+00
68	0	0.00E+00	2	1.08E-02	0	0.00E+00
69	0	0.00E+00	0	0.00E+00	0	0.00E+00

```
0     ** NEXT RANDOM NUMBER IS          0  4C5BA0D6
0TOTAL CPU TIME FOR THIS PROBLEM WAS 115.54 MINUTES.
1$$$$$$$$$$ Nova Upgrade Neutron-Gamma Effects Simulation 91
$$$$$$$$$$$$$$

 TODAY IS  1- 3-92
 END OF FILE READ BY INPUT1, LINE 42
 NORMAL COMPLETION OF JOB
```

Appendix E: MORSE Source Code

```
C       LMF.FOR
C**********************************************************************
C*        This version determines uncollided fluence and others.      *
C*        a collision-density estimator is used to determine          *
C*        fluence and is called for each collision.  This version     *
C*        includes SDATA for source neutrons, SGAM for neutron-       *
C*        generated gammas, and RELCOL for others.                    *
C**********************************************************************
C * * THIS IS THE MAIN ROUTINE * * * * * * * * * * * * * * * * * * * * *
C * * D. Beller July 89.  1 detector in LMF for X-ray effects
C * * THE FOLLOWING CARD DETERMINES THE SIZE ALLOWED FOR BLANK COMMON  *
c * * The value of NLFT below should be set to one less than this size
      COMMON NC(1000001)
C * *  (REGION SIZE NEEDED IS ABOUT 150K + 4*(SIZE OF BLANK COMMON IN WO
C * * NOTE - THE ORDER OF COMMONS IN THIS ROUTINE IS IMPORTANT AND MUST
C * *   POND TO THE ORDER USED IN DUMP ROUTINES SUCH AS HELP, XSCHLP, AN
C * *
C * * LABELLED COMMONS FOR WALK ROUTINES     * * * * * * * * * * * * * *
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(26),ITOUT,ITIN
      COMMON /FISBNK/ MFISTP
      COMMON /NUTRON/ NAME
C * *
C * * LABELLED COMMONS FOR CROSS-SECTION ROUTINES  * * * * * * * * * * *
      COMMON /LOCSIG/ ISCCOG
      COMMON /MEANS/ NM
      COMMON /MOMENT/ NMOM
      COMMON /QAL/ Q
      COMMON /RESULT/ POINT
C * *
C * * LABELLED COMMONS FOR GEOMETRY INTERFACE ROUTINES  * * * * * * * *
      COMMON /GEOMC/ XTWO
      COMMON /NORMAL/ UNORM
C * *
C * * LABELLED COMMONS FOR USER ROUTINES  * * * * * * * * * * * * * * *
      COMMON /PDET/ ND
      COMMON /USER/ AGST
C * *
C * * COMMON /DUMMY/ WILL NOT BE FOUND ELSEWHERE IN THE PROGRAM  * * * *
      COMMON /DUMMY/ DUM
C * *
        CHARACTER*40, NAM1
        CHARACTER*40, NAM2
        TYPE *,' '
        TYPE *,'********* MORSE Code, LMF X-Ray Effects Problem *****'
        TYPE *,'========> WARNING !!! <========'
        TYPE *,'ABORT if mixed x-secs are not assigned to FOR010'
        TYPE *,' '
        TYPE *,'ENTER NAME OF INPUT FILE'
```

```
        ACCEPT 100, NAM1
100     FORMAT(A40)
        TYPE *,'ENTER NAME OF OUTPUT FILE'
        ACCEPT 200, NAM2
200     FORMAT (A40)
        OPEN(UNIT=1,NAME=NAM1,TYPE='OLD')
        OPEN(UNIT=2,NAME=NAM2,TYPE='NEW')
      ITOUT = 2
      ITIN = 1
        NLFT=1000000
      CALL MORSE(NLFT)
        TYPE 300, NAM2
300     FORMAT(X,'OUTPUT FILE IS ',A40)
      STOP
      END
      SUBROUTINE GTMED(MDGEOM,MDXSEC)
C   FOR SETTING CROSS SECTIONS IN THE INPUT DATA FILE FOR MORSE
c     IF(MDGEOM.GT.0 .AND. MDGEOM.LT.1000) MDXSEC = MDGEOM
      MDXSEC = MDGEOM
      RETURN
      END

      FUNCTION DIREC
      COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
     1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
c for pathlength stretching toward the detectors (in the +y direction)
      DATA XD,YD,ZD/0.,400.,0/
      DIST = SQRT((XD-X)**2+(YD-Y)**2+(ZD-Z)**2)
      DIREC = (U*(XD-X)+V*(YD-Y)+W*(ZD-Z))/DIST
C
      RETURN
      END
      SUBROUTINE BANKR(NBNKID)
C  DO NOT CALL EUCLID FROM BANKR(7)
      COMMON /APOLLO/ AGSTRT,DDF,DEADWT(5),ETA,ETATH,ETAUSD,UINP,VINP,
     1 WINP,WTSTRT,XSTRT,YSTRT,ZSTRT,TCUT,XTRA(10),
     2   I0,I1,MEDIA,IADJM,ISBIAS,ISOUR,ITERS,ITIME,ITSTR,LOCWTS,LOCFWL,
     3 LOCEPR,LOCNSC,LOCFSN,MAXGP,MAXTIM,MEDALB,MGPREG,MXREG,NALB,
     4 NDEAD(5),NEWNM,NGEOM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,
     5 NKCALC,NKILL,NLAST,NMEM,NMGP,NMOST,NMTG,NOLEAK,NORMF,NPAST,
     6 NPSCL(13),NQUIT,NSIGL,NSOUR,NSPLT,NSTRT,NXTRA(10)
      COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
     1 ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
      NBNK = NBNKID
      IF (NBNK) 100,100,140
  100 NBNK = NBNK + 5
      GO TO (104,103,102,101),NBNK
  101 CALL STRUN
C     CALL HELP(4HSTRU,1,1,1,1)
      RETURN
  102 NBAT = NITS - ITERS
      NSAVE = NMEM
```

```
      CALL STBTCH(NBAT)
C    NBAT IS THE BATCH NO. LESS ONE
      RETURN
  103 CALL NBATCH(NSAVE)
C    NSAVE IS THE NO. OF PARTICLES STARTED IN THE LAST BATCH
      RETURN
  104 CALL NRUN(NITS,NQUIT)
C    NITS IS THE NO. OF BATCHES COMPLETED IN THE RUN JUST COMPLETED
C    NQUIT .GT. 1 IF MORE RUNS REMAIN
C          .EQ. 1 IF THE LAST SCHEDULED RUN HAS BEEN COMPLETED
C          IS THE NEGATIVE OF THE NO. OF COMPLETE RUNS, WHEN AN
C                EXECUTION TIME KILL OCCURS
      RETURN
  140 GO TO (1,2,3,4,5,6,7,8,9,10,11,12,13),NBNK
C   NBNKID   COLL TYPE   BANKR CALL       NBNKID   COLL TYPE   BANKR CALL
C      1       SOURCE     YES (MSOUR)        2       SPLIT       NO   (TESTW)
C      3       FISSION    YES (FPROB)        4       GAMGEN      YES (GSTORE
C      5       REAL COLL  YES (MORSE)        6       ALBEDO      YES (MORSE)
C      7       BDRYX      YES (NXTCOL)       8       ESCAPE      YES (NXTCOL
C      9       E-CUT      NO  (MORSE)       10       TIME KILL   NO   (MORSE)
C     11       R R KILL   NO  (TESTW)       12       R R SURV    NO   (TESTW)
C     13       GAMLOST    NO  (GSTORE)
    1 CALL SDATA
    2 RETURN
    3 RETURN
    4 Call SGAM
      Return
    5 CALL RELCOL
      RETURN
    6 RETURN
    7 RETURN
    8 RETURN
    9 RETURN
   10 RETURN
   11 RETURN
   12 RETURN
   13 RETURN
      END

      SUBROUTINE SDATA
c
c this version D. Beller, 12 July 1989, for an isotropic point
c source located at (X,Y,Z) and for point detectors 1 to ND located
c at (XD,YD,ZD).  Cos of angle is not stored!!!
c
      COMMON /USER/ AGSTRT,WTSTRT,XSTRT,YSTRT,ZSTRT,DFF,EBOTN,EBOTG,
1 TCUT,I0,I1,IADJM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,NLAST,
2 NLEFT,NMGP,NMTG,NSTRT
   COMMON /PDET/ ND,NNE,NE,NT,NA,NRESP,NEX,NEXND,NEND,NDNR,NTNR,NTNE,
1 NANE,NTNDNR,NTNEND,NANEND,LOCRSP,LOCXD,LOCIB,LOCCO,LOCT,LOCUD,
2 LOCSD,LOCQE,LOCQT,LOCQTE,LOCQAE,LMAX,EFIRST,EGTOP              SDATA 32
   COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD VOLD
```

```
1  ,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
     COMMON EN(1)                                               SDATA 50
        ENEST = 1.0
        DO 5 I=1,ND                                             SDATA 80
        ID = LOCXD + I
        XE = EN(ID)
        YE = EN(ID + ND)
        ZE = EN(ID + 2*ND)
        A = XE - X
        B = YE - Y
        C = ZE - Z
        SD2 = A*A + B*B + C*C
        SD = SQRT(SD2)
c diagnostic
        if(name.eq.1) type *, 'detector', I,'source-detector distance = ',
SD
c comment this out if sdata is working well
        TA = SD/EN(NMTG + IG) + AGE
        MARK = 1
        CALL EUCLID(MARK,X,Y,Z,XE,YE,ZE,SD,IG,ARG,0,NMED,BLZNT,NREG)
        IF (ARG.LT.-32) GO TO 5
        CON = WATE*EXP(ARG)/12.56637/SD2/ENEST
c          if (con.ge.0) goto 555
c          type *,'consdata =',con
        COS=B/sd                                                SGAM 250
555       CALL FLUXST(I,IG,CON,TA,cos,1)
5         Continue
C  *  *  SWITCH = -1  -- STORE IN ARRAY UD ONLY
c  *  *             1  -- Store in array UD and others          SDAT 150
        RETURN                                                  SDAT 160
        END                                                     SDAT 170

        SUBROUTINE SGAM                                         SGAM  10
C   Added for LMF problem 6 Feb 90 by D. Beller
* * * *
C   THIS VERSION ~S FOR POINT DETECTORS LOCATED AT (XD,YD,ZD)
* * * *
C    AND FOR AN ISOTROPIC POINT SOURCE
* * * *
C
* * * *
        COMMON /USER/ AGSTRT,WTSTRT,XSTRT,YSTRT,ZSTRT,DFF,EBOTN,EBOTG,
1 TCUT,I0,I1,IADJM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,NLAST,
      2 NLEFT,NMGP,NMTG,NSTRT                                   SGAM  62
        COMMON /PDET/
ND,NNE,NE,NT,NA,NRESP,NEX,NEXND,NEND,NDNR,NTNR,NTNE,SGAM  70
      1 NANE,NTNDNR,NTNEND,NANEND,LOCRSP,LOCXD,LOCIB,LOCCO,LOCT,LOCUD,
      2  LOCSD,LOCQE,LOCQT,LOCQTE,LOCQAE,LMAX,EFIRST,EGTOP
SGAM  72
        COMMON /NUTRON/ NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD
        1
,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
```

```
      COMMON EN(1)                                               SGAM  90
      DO 5 I=1,ND                                                SGAM 100
      ID = LOCXD + I                                             SGAM 110
      XE = EN(ID)                                                SGAM 120
      ID = ID + ND                                               SGAM 130
      YE = EN(ID)                                                SGAM 140
      ID = ID + ND                                               SGAM 150
      ZE = EN(ID)                                                SGAM 160
      A=X -XE                                                    SGAM 170
      B=Y -YE                                                    SGAM 180
      C=Z -ZE                                                    SGAM 190
      SD2=A*A+B*B+C*C                                            SGAM 200
      DS = SQRT(SD2)                                             SGAM 210
      TA = DS/EN(NMTG+IG)+AGE                                    SGAM 220
C * * * COS    DEPENDS ON THE ANGLE OF INTEREST * * *
      MARK = 1                                                   SGAM 260
      MEDIUM=NMED                                                SGAM 270
      CALL EUCLID(MARK,X,Y,Z,XE,YE,ZE,DS,IG,ARG,0  ,MEDIUM,BLZNT,NREG)
SGAM 280
      if (arg.lt.-64) goto 5
      CON = WATE *EXP(ARG)/12.56637/SD2                          SGAM 290
C  *   *  SWITCH =  1  -- STORE IN ALL RELEVANT ARRAYS            SGAM 300
C next two lines for current info
      COS=B/DS                                                   SGAM 250
      CALL FLUXST(I,IG,CON,TA,COS,1)
c     CALL FLUXST(I,IG,CON,TA, 1.0 ,1)                           SGAM 310
 5    CONTINUE                                                   SGAM 320
      RETURN                                                     SGAM 330
      END                                                        SGAM 340

      SUBROUTINE RELCOL                                          RELCO 10
C                                                                RELCO 20
C   THIS VERSION IS FOR POINT DETECTORS LOCATED AT (XD,YD,ZD) RELCO 30
C
                  RELCO 40
      COMMON /USER/ AGSTRT,WTSTRT,XSTRT,YSTRT,ZSTRT,DFF,EBOTN,EBOTG,
                                                                 RELCO 50
     1 TCUT,I0,I1,IADJM,NGPQT1,NGPQT2,NGPQT3,NGPQTG,NGPQTN,NITS,NLAST,
                                                                 RELCO 51
     2 NLEFT,NMGP,NMTG,NSTRT                                     RELCO 52
      COMMON /PDET/
ND,NNE,NE,NT,NA,NRESP,NEX,NEXND,NEND,NDNR,NTNR,NTNE,             RELCO 60
     1 NANE,NTNDNR,NTNEND,NANEND,LOCRSP,LOCXD,LOCIB,LOCCO,LOCT,LOCUD,
                                                                 RELCO 61
     2 LOCSD,LOCQE,LOCQT,LOCQTE,LOCQAE,LMAX,EFIRST,EGTOP         RELCO 62
      COMMON /NUTRON/
NAME,NAMEX,IG,IGO,NMED,MEDOLD,NREG,U,V,W,UOLD,VOLD               RELCO 70
     1
,WOLD,X,Y,Z,XOLD,YOLD,ZOLD,WATE,OLDWT,WTBC,BLZNT,BLZON,AGE,OLDAGE
                                                          RELCO 71
      COMMON BL(1)                                               RELCO 80
      DIMENSION NL(1)                                            RELCO 90
```

```
      EQUIVALENCE (BL(1),NL(1))                                   RELC 100
      DATA NEST /2/, FNEST /2./
C                                    above is                     RELC 110
C   NEST + FNEST ARE THE NO. OF ESTIMATES TO BE MADE TO EACH DETECTOR
                                                                  RELC 130
C * * * ISTAT MUST BE EQUAL TO 1.                                        *
* * *
C * * * NEX MUST BE AT LEAST 1
* * * *
C * * * NEXND MUST BE AT LEAST 1                                        *
* * *
      DO 30 I=1,ND                                                RELC 160
      IA=LOCXD+I                                                  RELC 170
      XE = BL(IA)                                                 RELC 180
      YE = BL(IA+ND)                                              RELC 190
      ZE = BL(IA+2*ND)                                            RELC 200
      A = XE - X                                                  RELC 210
      B = YE - Y                                                  RELC 220
      C = ZE - Z                                                  RELC 230
      SD2=A*A+B*B+C*C                                             RELC 240
      DS=SQRT (SD2)                                               RELC 250
C * * * COS DEPENDS ON THE ANGLE OF INTEREST
* * * *
      COS=B/DS                                                    RELC 270
      THETA = (A*UOLD + B*VOLD + C*WOLD)/DS                       RELC 280
      IGOLD = IGO                                                 RELC 290
      IGQ = NGPQT3                                                RELC 300
      IF (IGO.LE.NGPQT1) IGQ=NGPQT1                               RELC 310
      IA = LOCRSP + NRESP*NMTG + 1                                RELC 320
      CALL PTHETA(NMED,IGOLD,IGQ,THETA,BL(IA),NMTG)               RELC 330
      NES = 0                                                     RELC 340
      PSUM = 0.                                                   RELC 350
      IA = IA - 1                                                 RELC 360
      DO 5 IL=IGOLD,IGQ                                           RELC 370
 5    PSUM = PSUM +  (BL(IA+IL))                                  RELC 380
C samples from a normalized distr without negative Legendre coeffs
removed
 10   R = FLTRNF(0) * PSUM                                        RELC 390
      DO 15 IL=IGOLD,IGQ                                          RELC 400
      if (bl(ia+il).lt.0) goto 15
      if (r .lt. 0) goto 15
      IF (R -  (BL(IA+IL))) 20,20,15                              RELC 410
 15   R = R -  (BL(IA+IL))                                        RELC 420
      IL = IGQ                                                    RELC 430
 20   MARK=1                                                      RELC 440
      AGED = AGE + DS/BL(NMTG+IL)                                 RELC 450
      MEDIUM=NMED                                                 RELC 470
      CALL EUCLID(MARK,X,Y,Z,XE,YE,ZE,DS,IL,ARG,0,MEDIUM,BLZNT,NREG)
RELC 480
      IF (ARG.LT.-64.) GO TO 25                                   RELC 490
C********BEWARE THIS VERSION WILL NOT WORK IF ENERGY BIASING IS USED
* * * *
```

```
      CON = WATE*EXP (ARG)*SIGN (PSUM,BL(IA+IL))/SD2/FNEST  RELC 510
C * * * couldn't handle 1e-40
      IF (CON.LT.1.0E-36) GO TO 25                          RELC 520
c      type *,'con=',con,'  group = ',il,'  wate = ',wate,
c     1'  exp = ',exp(arg),'bl(ia+il) = ',bl(ia+il),'distance = ',ds
      CALL FLUXST (I,IL,CON,AGED,COS,0)                     RELC 530
 25   NES = NES + 1                                         RELC 540
      INN=LOCXD+6*ND+I                                      RELC 550
      NL(INN)=NL(INN)+1                                     RELC 560
      IF (NES-NEST) 10,30,30                                RELC 570
 30   CONTINUE                                              RELC 580
      RETURN                                                RELC 590
      END                                                   RELC 600
```

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13. ABSTRACT (*Maximum 200 words*)

Gamma ray effects testing in Lawrence Livermore National Laboratory's (LLNL) planned Nova Upgrade facility is examined. Emphasis is placed on converting neutron energy from inertial confinement fusion in to gamma rays while shielding the test objects from neutrons and debris. Although predicted gamma doses in the Nova Upgrade facility are an order of magnitude less than those produced in some current facilities, dose uniformity, the ratio of minimum to maximum gamma dose is predicted to be greater than 0.75 across a larger, 13,000 cm^2, test bed. Peak gamma dose rates are predicted to be on the order of 10^{10} Gy/s, similar to the dose rates of current simulators. Surprisingly, the laser ports reduce the gamma dose about 30% and the peak gamma dose rate about 40%, but they increase the average gamma energy about 20%. The dose and dose rates from the Nova nuclear weapons effects test (NWET) cassette should scale linearly with the yield from future ICF facilities, such as the Laboratory Microfusion Facility (LMF) planned by the Department of Energy.

14. SUBJECT TERMS
Nuclear Weapons Effects Testing, Gamma-Ray Effects Testing, Fusion, Inertial Confinement Fusion, Gamma Rays, Neutrons

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